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LIST OF ABBREVIATIONS, TERMS AND UNITS

AC	Alternating Current
ALARA	As Low As is Reasonably Achievable
ANSI	American National Standards Institute
AOR	Abnormal Occurrence Report
ASME	American Society of Mechanical Engineers
BOP	Balance of Plant
CFH	Certified Fuel Handler
CFR	Code of Federal Regulations
Ci	Curie; Unit of Radioactivity = 3.7E-10 Disintegrations Per Second
CRP	Component Removal Project
DAC	Derived Air Concentration
DC	Direct Current
DECON	Immediate Decontamination and Dismantlement Option
DOE	Department of Energy
DPM	Disintegrations per Minute
EDM	Electrical Discharge Machining
ENTOMB	Encasement in Concrete with Future Dismantlement Option
EPA	Environmental Protection Agency
FERC	Federal Energy Regulatory Commission
FSAR	Final Safety Analysis Report
FTE	Fuel Transfer Enclosure
GET	General Employee Training
GM	Geiger-Mueller
HEPA	High Efficiency Particulate Air (Filter)
ICRP	International Commission on Radiological Protection
INPO	Institute of Nuclear Power Operators
ISFSI	Independent Spent Fuel Storage Installation
ISR	Independent Safety Review
kV	Kilovolt
LLD	Lower Limit of Detection
LLW	Low Level Waste
LPG	Liquid Propane Gas
M&TE	Measuring and Test Equipment
μCi	0.000001 Ci
MCS	Main Coolant System
MDA	Minimum Detectable Activity
MDC	Minimum Detectable Concentration
MDM	Metal Disintegration Machining
mR	0.001 R
μR	0.000001 R
MWe	Megawatts Electric

MWt	Megawatts Thermal
NCRP	National Council on Radiation Protection and Measurements
NIST	National Institute of Standards and Technology
NPDES	National Pollutant Discharge Elimination System
NRC	Nuclear Regulatory Commission
NSARC	Nuclear Safety Audit and Review Committee
NSSS	Nuclear Steam Supply System
NST	Neutron Shield Tank
ODCM	Off-Site Dose Calculation Manual
OSHA	Occupational Safety and Health Administration
PAB	Primary Auxilary Building
PAG	Protective Action Guides
PASS	Post-Accident Sample System
PCA	Potentially Contaminated Area
PCB	Polychlorinated Biphenyls
pCi	0.000000000001 Ci
Person-rem	Collective Radiation Dose to a Population
PIC	Pressurized Ion Chamber
PIR	Plant Information Report
POL	Possession Only License
PORC	Plant Operation Review Committee
PSDAR	Post-Shutdown Decommissioning Activities Report
R	Roentgen; Unit of Radiation Exposure
RCA	Radiation Control Area
RCRA	Resource Conservation and Recovery Act
Rem	Unit of Dose Equivalent
REMP	Radiological Environmental Monitoring Program
RETS	Radiological Effluent Technical Specifications
RWP	Radiation Work Permit
SAFSTOR	Delayed Decontamination and Dismantlement Option
SFP	Spent Fuel Pit
TEDE	Total Effective Dose Equivalent
TLD	Thermoluminescent Dosimeter
TLG	TLG Engineering, Inc.
TS	Technical Specifications
VC	Vapor Container
WD	Waste Disposal
YDQAP	Yankee Decommissioning Quality Assurance Program
YAEC	Yankee Atomic Electric Company
YNPS	Yankee Nuclear Power Station

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YANKEE NUCLEAR POWER STATION (YNPS)

POST-SHUTDOWN DECOMMISSIONING ACTIVITIES REPORT (PSDAR)

(As of June 2001)

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INTRODUCTION AND OVERVIEW OF THE PSDAR

Pursuant to 10CFR50.82(a)(4)(i) and Regulatory Guide 1.185, this appendix contains information concerning post-shutdown activities remaining at the Yankee Nuclear Power Station (YNPS). Section 50.82(a)(4)(i) requires that licensees develop a post-shutdown decommissioning activities report (PSDAR). Licensees with an approved decommissioning plan, however, may "replace their decommissioning plans with a PSDAR update that uses the format and content specified in this document" (Reference 13). The YNPS Decommissioning Plan (Reference 1), which was approved on February 14, 1995 and later became part of the Final Safety Analysis Report (FSAR), describes all remaining decommissioning activities, but in considerably more detail than that required in the PSDAR. Yankee Atomic Electric Company (YAEC) has elected to relocate pertinent information to a PSDAR, which conforms to the guidance in RG 1.185.

HISTORICAL BACKGROUND

YNPS achieved initial criticality in 1960 and began commercial operations in 1961. The Nuclear Steam Supply System is a four loop pressurized water reactor designed by Westinghouse Electric Corporation. The original thermal power design limit of 485 MWt was upgraded to 600 MWt in 1963. The Turbine Generator, also designed by Westinghouse, was rated to produce 185 MWe. Commercial operation ceased in 1992 after about 31 years of operation. During its operation, YNPS achieved an average capacity factor of about 74%.

YNPS shut down on October 1, 1991 in response to regulatory uncertainties associated with the integrity of the Reactor Vessel. During the outage and before February 26, 1992 all fuel assemblies, control rods, and neutron sources were removed from the Reactor Vessel and stored in the Spent Fuel Pit. A total of 533 fuel assemblies are stored in the Spent Fuel Pit. Plant systems required to support spent fuel storage and to support permanently defueled operations are in service.

On February 26, 1992, the YAEC Board of Directors decided to cease power operations permanently at YNPS. By letter, dated February 27, 1992 (Reference 2), YAEC notified the Nuclear Regulatory Commission (NRC) of the Company's decision to permanently cease power operations at the YNPS. After notifying the NRC, YAEC initiated decommissioning planning and other plant closure activities to safely reduce radioactivity at the site to residual levels, allowing release of the site for unrestricted access.

On August 5, 1992, the NRC amended the YNPS Facility Operating License (DPR-3) to possession only status (Reference 3). This, combined with other amendments and program changes, formed the basis of the Decommissioning Plan. The plan (Reference 1) was submitted by YAEC in accordance with the requirements of 10CFR50.82(a) [pre-1996], that required submittal of a proposed Decommissioning Plan within two years of the permanent cessation of operations. The Decommissioning Plan was subsequently approved on February 14, 1995 (Reference 4). A commitment from the approval process required that the Decommissioning Plan be incorporated into the FSAR.

After deciding to close YNPS permanently, YAEC reviewed the plant licensing basis to evaluate the applicability of existing Technical Specifications and NRC regulations to a permanently defueled condition. The Technical Specifications that were applicable only to an operating mode

corresponding to fuel in the reactor were deemed inapplicable to the permanently defueled condition. Section 504 of the FSAR summarizes the Technical Specifications that will be in effect during decommissioning.

Plant closure activities, which were commenced following the decision to cease power operations, will continue in accordance with applicable regulatory requirements and YAEC's commitment to maintain the facility in a safe and economical manner. These activities have included site security modifications, control rod disposal, decontamination, disposal of radioactive components, lay-up of plant equipment, and facility modifications to improve plant operations.

DECOMMISSIONING ALTERNATIVE

Following an evaluation of the three decommissioning alternatives, YAEC selected DECON as the most appropriate alternative for decommissioning YNPS. YAEC's choice of a decommissioning alternative is consistent with NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities,"(Reference 6). In that document, the NRC concluded that DECON and SAFSTOR alternatives are reasonable for decommissioning a pressurized water reactor. Implementation of the DECON alternative is within the 60 year limit (after cessation of operation) in 10CFR50.82(a)(3).

DESCRIPTION OF DECOMMISSIONING ACTIVITIES

Since 1993 YAEC has removed and disposed of the steam generators, pressurizer, reactor vessel and reactor vessel internals. The reactor vessel internals components, which are Greater-Than-Class-C (GTCC) waste, remains onsite and are stored with the spent fuel in the spent fuel pool (SFP). YAEC has sought accelerated acceptance of its spent fuel by the Department of Energy (DOE) in accordance with the current fuel disposal contract. The DOE's position is that they have not yet determined whether priority will be accorded shutdown reactors, or if priority is granted, under what specific circumstances it might be granted.

As of June 2001, the majority of systems and components not required to support the storage of spent fuel have been dismantled and disposed of in accordance with the YNPS Decommissioning Plan and Final Safety Analysis Report (References 1 and 12). In addition, the Spent Fuel Pool and other systems associated with fuel storage have been electrically and mechanically isolated to create a Spent Fuel Pool "Island" that will not be adversely impacted by ongoing decommissioning activities. The current status of the systems, structures, and components are described in Table-1. The purpose of this section is to identify the dismantlement and decontamination activities associated with the remaining systems, structures, and components at YNPS.

The remaining decommissioning activities will be completed in three phases. The current phase consists of the removal of any remaining systems and components that do not support fuel storage or subsequent decontamination activities. After the removal of all spent fuel and GTCC waste from the SFP, the second phase of decommissioning will involve the dismantlement and decontamination of the SFP and its supporting systems, structures, and components. In the final phase of decommissioning, the possession only license will be terminated.

A. Remaining Systems, Structures, And Components Not Necessary For Spent Fuel Storage (Phase 1)

After removing systems and components from an area or building, contaminated concrete, steel, and other building materials will be decontaminated or removed. The structures listed below are not required to store spent fuel and could be decontaminated and decommissioned.

- Vapor Container
- Reactor Support Structure
- Upper and Lower Pipe Chases
- Fuel Transfer Chute
- South, East, and West Walls of Ion Exchange Pit
- Primary Auxiliary Building
- Waste Disposal Building
- Safe Shutdown Building
- Potentially Contaminated Area (PCA) Buildings 1 and 2
- Compactor Building
- Turbine Building
- Service Building
- Storage Warehouse
- Meteorological Tower

Following License Termination Plan approval, final status surveys will be conducted to verify that structures and open land areas not required for spent fuel storage meet the release criteria. Independent verification of the results by the NRC will allow for the release of the individual surveyed structures and open land areas as non-radiologically controlled material available for conventional demolition and disposal. In order to facilitate remediation, the facility superstructures may be released and demolished prior to remediating substructure and soils work beneath the structures. Measures will be implemented to prevent recontamination of surveyed areas prior to final license termination.

B. Systems, Structures, And Components Associated With Storage Of Spent Fuel In The Spent Fuel Pool (Phase 2)

After removing the spent fuel and GTCC waste from the SFP, the remaining components of the systems listed below will be dismantled and decontaminated.

- Radiation Monitoring System
- Fuel Handling System
- SFP Cooling and Purification System
- Temporary Liquid Waste Processing System
- Auxiliary Service Water System
- Demineralized Water System
- Electrical System
- Ventilation System

- Fire Protection and Detection System

After the removal of systems and components from an area or building associated with the SFP Island, contaminated structural concrete, steel and other building materials will be decontaminated or removed. The structures listed below will be decontaminated during decommissioning of the SFP Island.

- Yard Area Crane and Support Structure
- Ion Exchange Pit
- Primary Vent Stack
- Spent Fuel Pool and SFP Building
- New Fuel Vault
- Potentially Contaminated Area Warehouse
- Service Building

Following License Termination Plan approval, final status surveys will be conducted to verify that structures and open land areas associated with the SFP meet the release criteria. Independent verification of the results by the NRC will allow for the release of the individual surveyed structures and open land areas as non-radiologically controlled material available for conventional demolition and disposal. In order to facilitate remediation, the superstructures may be released and demolished prior to remediating substructure and soils work beneath the structures. Measures will be implemented to prevent recontamination of surveyed areas prior to final license termination.

C. License Termination (Phase 3)

The final phase of decommissioning will take place after all spent fuel and GTCC waste is taken off-site. In the interim, spent fuel and GTCC waste may be stored either in the Spent Fuel Pool or in an on-site dry cask storage facility. If the spent fuel and GTCC waste is kept in the SFP until it is taken off-site, license termination will take place immediately after decontamination and dismantlement of the SFP island. If a dry cask storage facility is constructed and operated, the final phase of decommissioning will follow the dismantlement and decontamination of this facility.

Decommissioning of the dry cask storage facility would consist primarily of the disposal of the concrete canister overpacks, provided they are not shipped with the spent fuel casks. The overpack design minimizes neutron activation, thereby generating minimal radioactive waste. This waste should qualify for disposal at a low-level radioactive waste disposal site.

Regardless of the spent fuel storage option chosen, the 10CFRPart 50 license will be terminated after the last stage of final status surveys and independent NRC verification. Site restoration activities will then be conducted to return the remaining small portion of the site to a “green field” condition.

OTHER DECOMMISSIONING CONSIDERATIONS

The dismantlement and, if necessary, decontamination of contaminated systems, structures and components may be accomplished using the following three approaches: decontamination in place, dismantlement and decontamination, or dismantlement and disposal. Furthermore, a combination of these methods may reduce contamination levels, worker radiation exposures, and project costs. General considerations applicable to these activities are described in detail in Section 200 of the FSAR and are summarized below.

A. General Decommissioning Activities Relating to Removal of Radiological Components & Structures

Components will be safely and efficiently removed using the techniques and appropriate methods for the particular circumstances and consistent with Decommissioning Work Packages. Openings in components will typically be covered and sealed to minimize the spread of contamination. The components may be moved to a processing area for volume reduction and packaging into containers for shipment to a processing facility for decontamination or a low-level radioactive waste disposal facility. Buried contaminated components (e.g., piping, drains, etc.) will be decontaminated in place or excavated.

B. Decontamination Methods

Contaminated systems and components will be removed and sent to an off-site processing facility or to a low level radioactive waste disposal facility. On-site decontamination of systems and components will generally be limited to activities needed to maintain personnel exposure as low as is reasonably achievable, to expedite equipment removal, and to control the spread of contamination.

Application of coatings and hand wiping will be the preferred methods for stabilizing or removing loose surface contamination. If other methods are utilized (e.g., grit blasting, high pressure water), airborne contamination control and waste processing systems will be used as necessary to control and monitor any releases of contamination.

Contaminated and activated concrete as well as other contaminated materials will be removed and sent to a low-level radioactive waste disposal facility. Concrete removal methods, such as scabbling and scarifying, will control the removal depth to minimize the waste volume produced. Vacuuming the dust and debris effluent with HEPA filtration will minimize the need for additional respiratory protection control measures. While these methods represent the most practicable and widely used decontamination methods available at this time, YAEC will consider new decontamination technologies if appropriate.

C. Dismantlement Methods

YAEC intends to use two basic dismantlement methods:

- *Mechanical Methods* - Mechanical methods machine the surfaces of the material that is being cut. Typically, these methods are capable of cutting remotely without generating

significant amounts of airborne contamination. This attribute makes these methods attractive for removing most of the contaminated piping, equipment, and components at YNPS. The outside diameter machining method, in particular, is best suited for cutting large bore contaminated piping. Smaller bore contaminated piping, tubing, and supports can be cut using any of the mechanical methods (e.g., band saws, reciprocating saws, hydraulic shears).

- *Thermal Methods* - Thermal methods melt or vaporize the surfaces of material. The cutting debris is transported from the cut region with a gas jet or water spray. Although thermal methods are significantly quicker than mechanical methods, they have high power requirements and generate airborne contamination when used on contaminated systems in air. Generation of airborne contamination can be easily controlled when the method is used underwater. Thermal methods are suitable for segmenting large vessels in areas that can easily be sealed, filtered, or maintained underwater. The method is also suitable for use at a cutting station with air filtration. Thermal methods are appropriate for removing structural steel if it has been decontaminated or if a local contamination envelope with HEPA filtration is established. Appropriate lead paint removal controls must also be implemented.

While these methods represent the most practicable and widely used decontamination methods available at this time, YAEC will consider new decontamination technologies if appropriate.

D. Special or Unusual Programs

There are no special or unusual programs. All procedures and processes that will be used at YNPS are consistent with those considered in the Final Generic Environmental Impact Statement (FGEIS).

E. Removal of Low Level Radioactive Waste (LLW) and Compaction or Incineration

LLW will be processed in accordance with plant procedures and sent to LLW disposal facilities. While no incineration will occur onsite, YAEC may use an off-site licensed facility. YAEC has no intention, however, to use onsite compaction at this time.

F. Soil Remediation

Soils and pavement will be surveyed and characterized in accordance with the site radiological characterization program. As necessary, soils, and pavement will be remediated (i.e., removed, processed and disposed of at a licensed facility) if determined to contain contamination levels above the guideline values established in an approved License Termination Plan.

G. Processing and Disposal Site Locations

Currently, there are several facilities available for (1) the processing of waste materials to achieve volume reduction prior to disposal or (2) the disposal of low level radioactive waste.

These locations include (but are not limited to) Chem Nuclear - Barnwell, South Carolina; Envirocare - South Clive, Utah; GTS Duratek - Oak Ridge, Tennessee.

H. Removal of Mixed Wastes

Mixed wastes will be managed according to all applicable federal and state regulations including NRC handling, storage, and transportation regulations. Mixed wastes from YNPS will be transported only by authorized and licensed transporters and shipped only to authorized and licensed facilities. If technology, resources, and approved processes become available, processes will be evaluated to render the mixed waste non-hazardous.

I. Storage/Removal of Spent Fuel and GTCC Waste

Beginning in 2001, YAEC plans to store spent fuel and GTCC waste in an Independent Spent Fuel Storage Installation (ISFSI), until the DOE is ready to take such waste. YAEC cannot make a precise determination of when spent fuel and GTCC waste will be removed from the YNPS because the availability of a licensed DOE high level waste repository is uncertain. Currently, YAEC expects that the turnover of spent fuel and GTCC waste to the DOE will be completed in 2020.

J. License Termination Plan, Final Radiological Survey and Site Release Criteria

The ultimate goal of decommissioning YNPS is to release the site for unrestricted use. This requires assurance that future uses of the site, after license termination, will not expose members of the general public to unacceptable levels of radiation.

In May 1997, YAEC submitted to the Commission for approval a License Termination Plan (LTP) for YNPS, pursuant to 10CFR50.82, as amended by 62 Fed. Reg. 39091 (July 29, 1996). YAEC's LTP employed a survey methodology based on the "Manual for Conducting Radiological Surveys in Support of License Termination (Reference 9)," known as the NUREG-5849 Methodology. Subsequently, the Commission, jointly with a group of other federal agencies, approved an alternative survey methodology, known as NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual" (MARSSIM – Reference 10). In May 1999 (Reference 11), YAEC advised the Commission that it intended to shift from the NUREG-5849 Methodology to MARSSIM and withdrew its previously submitted LTP application. YAEC will submit a new LTP at some time in the future, but no schedule has been set. The new plan will conform to NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," April 2000, and will address the limits of 10CFR20. Final status surveys will then be conducted to verify that structures and open land areas meet the release criteria. Finally, an independent NRC contractor will conduct a verification survey, thereby allowing unrestricted release of the site. After final status surveys and NRC verification, individual surveyed structures and open land areas will be released as non-radiologically controlled material for conventional demolition and disposal. YAEC will, nevertheless, maintain control over the site until termination of its 10CFRPart 50 license.

K. Site Restoration

Following termination of the YNPS possession only license by the NRC, YAEC will commence site restoration activities. Some of these activities, however, may be completed during the dismantlement period. Activities associated with the Vapor Container will include removal of internal structures, disassembly of the Vapor Container shell, and demolition of the Reactor Support Structure and other concrete structures. Instrumentation will be removed from the Control Room and other remote control areas once the instrumentation no longer supports plant activities. The remaining plant structures will be demolished. All building foundations will be back filled with concrete and structural fill. Site areas will be graded and landscaped as necessary to restore the site to a "green field" condition.

SCHEDULE OF DECOMMISSIONING ACTIVITIES

As stated above, decommissioning will be completed in three phases. The current phase consists of the decontamination and dismantlement of remaining systems and components that do not support fuel storage. After the removal of all spent fuel from the SFP, the second phase of decommissioning will involve the dismantlement and decontamination of the SFP and its supporting systems, structures and components. The final phase of decommissioning will involve the termination of the possession only license. License termination will occur after all spent fuel has been taken off-site. All decommissioning activities will be accomplished with no significant adverse environmental impacts.

Figure 1-1 shows the current schedule for the activities described above. YAEC expects to complete the current phase of dismantlement and decontamination and final status surveys by early 2003. The design and construction of a dry cask storage facility is expected to be completed in 2001. Following the transfer of spent fuel and GTCC waste from the SFP, decommissioning of the SFP island will begin and last approximately two years, including final status surveys. The dry cask storage facility is expected to be operated from 2001 to 2020, when the last fuel assembly is assumed to be taken off-site. Using this assumption, the YNPS license will be terminated between 2021 and 2022 after the dry cask storage facility is decommissioned.

Planning sequences and dates are based on current knowledge and could change in the future. Several factors influenced the choice of the spent fuel storage method at YNPS including the availability of a low-level radioactive waste site; the impact of the chosen method on decommissioning the balance of the site; the timing of spent fuel acceptance by the DOE; and the relative economics of dry versus wet storage of spent fuel. Although unlikely, Yankee may store spent fuel and GTCC in the SFP through the remainder of decommissioning. Both wet and dry storage options are addressed in the spent fuel management plans contained in the FSAR. Yankee will continue to inform the NRC of all major changes to the planned decommissioning activities in accordance with 10CFR50.82(a)(7).

DECOMMISSIONING COST ESTIMATE

The YNPS Decommissioning Plan (Reference 1) was submitted to the NRC in December 1993 and included a cost study for operating the facility through a safe storage period, decommissioning the facility, restoring the site to a "green field," and storing spent fuel until its transfer to the DOE. In

October 1994, Yankee completed a revised cost study to assist the NRC in its review of the Decommissioning Plan and to fulfill a commitment to Federal Energy Regulatory Commission (FERC). This 1994 cost study was based on the assumption that dismantlement activities would not begin until a low-level radioactive waste disposal site became available to Yankee in 2003.

In June 1995, the State of South Carolina re-opened the low-level waste facility in Barnwell, South Carolina to radioactive waste generators throughout the United States. In response, Yankee updated the cost estimate to reflect several significant changes in parameters affecting decommissioning costs. This study, called the 1995 Cost Study, was filed with FERC in August 1995. This study was a site-specific cost study that adjusted the 1994 Cost Study for differences in decommissioning timing, waste disposal costs, and one year of escalation. The 1995 Cost Study estimate of “to-go” costs remaining as of January 1995 was $303.2 million. In addition, as part of the final December 1995 FERC settlement, Yankee was allowed to collect another $3.2 million in the decommissioning trust fund to adjust for adjudicatory delays during re-approval of the Decommissioning Plan, bringing the total January 1995 “to-go” cost to $306.4 million (1995 dollars).

As required by the FERC settlement, an updated cost estimate was filed in December 1999. The total decommissioning costs remaining (as of January 1, 1999) are $246 million in constant value 1999 dollars. The total cost of decommissioning (expressed in 1999 dollars) is summarized below:

• Expended dollars (1993 – 1998)	$207,100,000
• Dismantlement Activities	147,700,000
• Spent Fuel Storage	98,300,000
• **Total Cost to Remove/Dismantle YNPS**	**$453,100,000**

Yankee has collected its decommissioning funds through its Power Contracts. The collections were deposited in an independent and irrevocable trust at a commercial bank, with the principal and interest used to discharge decommissioning obligations as they are incurred. This trust is in compliance with 10CFR50.75(e)(1)(ii) and a copy of the trust document has been provided to the NRC. The Power Contracts obligate the purchasers for the full costs of decommissioning YNPS, including spent fuel. The FERC orders received by Yankee acknowledge the continuing obligation of the purchasers with respect to the full cost of decommissioning YNPS. The periodic reviews of decommissioning cost studies mandated by FERC provide the mechanism for updating the required payments under the Power Contracts to assure adequate funds for that purpose.

ENVIRONMENTAL IMPACTS

YAEC prepared an Environmental Report [Reference 5] to evaluate all actual or potential environmental impacts associated with the proposed decommissioning activities. This evaluation used as its basis NUREG-0586, “Final Generic Environmental Impact Statement (FGEIS) on Decommissioning of Nuclear Facilities,” [Reference 6] and the site-specific environmental assessment from the re-capture of the construction period time duration [Reference 7].

This Environmental Report concluded that the impacts due to decommissioning of the YNPS will be bounded by the previously issued environmental impact statements, specifically the FGEIS and previously issued environmental assessments. This is principally due to the following reasons:

- The postulated impacts associated with the method chosen, DECON, have already been considered in the FGEIS.

- There are no unique aspects of the plant or decommissioning techniques to be utilized that would invalidate the conclusions reached in the FGEIS.

- The methods to be employed to dismantle and decontaminate the site are standard construction based techniques fully considered in the FGEIS.

- The site-specific person-rem estimate for all decommissioning activities has been conservatively calculated using methods similar to and consistent with the FGEIS.

Specifically, this review concludes that the YAEC decommissioning will result in generally positive environmental effects, in that:

- Radiological sources that create the potential for radiation exposure to site workers and the public will be eliminated.

- The site will be returned to a condition that will be acceptable for unrestricted use.

- The thermal impact on the Deerfield River from facility operations will be eliminated.

- Noise levels in the vicinity of the facility will be reduced.

- Hazardous materials and chemicals will be removed.

- Local traffic will be reduced (fewer employees, contractors and materials shipments than are required to support an operating nuclear power plant).

Furthermore, the YNPS decommissioning will be accomplished with no significant adverse environmental impacts in that:

- No site specific factors pertaining to YNPS would alter the conclusions of the FGEIS.

- Radiation dose to the public will be minimal.

- Radiation dose to decommissioning workers will be a fraction of the operating experience.

- Decommissioning is not an imminent health or safety problem and will generally have a positive environmental impact.

The total radiation exposure impact for decommissioning was estimated in the Decommissioning Plan (Reference 1) to be approximately 744 person-rem. This estimate was re-evaluated in 1996, resulting in a lower value of 580 person-rem (Reference 8). The actual exposure, through 12/31/00, for decommissioning activities is 529person-rem. The estimated radiation exposure for completion of the decommissioning activities at YNPS accounts for the one remaining dose intensive task of fuel transfer to dry cask and subsequent decontamination and dismantlement of the spent fuel pit.

This “to go” exposure has been estimated to be 43 person-rem (excludes public and transportation dose). Fuel transfer personnel exposure is based on a 1 person-rem/cask loaded and should be very conservative as current industry experience indicates that 0.5 person-rem/cask is achievable.

Radiation exposure due to transportation of radioactive waste has been conservatively estimated to be approximately 7 person-rem. This value is bounded by the FGEIS value of 100 person-rem for transportation occupational exposure.

Radiation exposure to off-site individuals for expected conditions, or from postulated accidents is bounded by the Environmental Protection Agency’s Protective Action Guidelines and NRC regulations. The public exposure due to radiological effluents will continue to remain well below the 10CFRPart 20 limits and the ALARA dose objectives of 10CFR50, Appendix I. This conclusion is supported by the YNPS Annual Effluent Release Reports in which individual doses to members of the public are calculated for station liquid and gaseous effluents.

No significant impacts are expected from the disposal of low-level radioactive waste (LLW). The total volume of YNPS low-level radioactive waste for disposal was estimated in the Decommissioning Plan to be approximately 132,000 ft^3. A review of the annual effluent reports filed with the NRC has determined that, through the end of 2000, 128,644 ft^3 of LLW from YNPS has been shipped off-site for burial. The waste volume that remains at YNPS is bounded by the FGEIS estimate of 647,670 ft^3 for a reference PWR.

Since the approval of the Decommissioning Plan (Reference 1) and the issuance of the Decommissioning Environmental Report (Reference 5), YNPS has identified the presence of solid Polychlorinated Biphenyl’s (PCB’s) in some paint coatings. As in the case of radiologically contaminated lead paint, asbestos, and other hazardous materials, contaminated paint that contains PCB’s will be managed according to all applicable federal and state regulations.

No significant environmental impacts are anticipated in the event that LLW is required to be temporarily stored onsite because adequate storage space exists and LLW storage will be in accordance with all applicable federal and state regulations.

The non-radiological environmental impacts from decommissioning are temporary and are not significant. The largest occupational risk associated with decommissioning YNPS is related to the risk of industrial accidents. The primary environmental effects are short term, small increases in noise levels and fugitive dust in the immediate vicinity of the site, as well as truck traffic to and from the site for hauling equipment and waste. No socioeconomic impacts, other than those associated with cessation of operation (loss of jobs and taxes), or impacts to local culture, terrestrial or aquatic resources such as the Sherman Reservoir and Deerfield River have been identified.

REFERENCES

1. Letter, J. K. Thayer (YAEC) to T. E. Murley (USNRC), Decommissioning Plan for Yankee Nuclear Power Station (YNPS), December 20, 1993.

2. Letter, A. C. Kadak (YAEC) to T. E. Murley (USNRC), Permanent Cessation of Power Operations at the Yankee Nuclear Power Station, BYR 92-024, February 27, 1992.

3. Letter, M. B. Fairtile (USNRC) to J. M. Grant (YAEC), Issuance of Amendment #142 to Facility License DPR-3 – Yankee Nuclear Power Station, August 5, 1992.

4. Letter, M. B. Fairtile to J. A. Kay (YAEC), Order Approving the Decommissioning Plan and Authorizing Decommissioning of the Yankee Nuclear Power Station, February 14,1995.

5. Yankee Nuclear Power Station Decommissioning Environmental Report, December 1993.

6. NUREG-0586, Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities, August 1988.

7. Letter, M. B. Fairtile (USNRC) to G. Papanic (YAEC), dated June 2, 1988.

8. USNRC Atomic Safety and Licensing Board Docket No. 50-029-DCOM, Supplemental Affidavit of Russell A. Mellor, September 3, 1996.

9. NUREG-5849, "Manual for Conducting Radiological Surveys in Support of License Termination," Draft, June 1992.

10. NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual," December 1997.

11. Letter, D.K. Davis (YAEC) to USNRC, "Withdrawal of Proposed License Amendment to Approve Yankee Nuclear Power Station's License Termination Plan," May 25, 1999.

12. Yankee Nuclear Power Station Final Safety Analysis Report.

13. Regulatory Guide 1.185, Standard Format and Content for Post-Shutdown Decommissioning Activities Report, July 2000.

TABLE–1: STATUS OF SYSTEMS, STRUCTURES AND COMPONENTS DESCRIBED IN FSAR

System, Structure, or Component	FSAR Section	Needed during Wet Fuel Storage	Status
Reactor Vessel	201	No	Removed.
Steam Generators	202	No	Removed.
Main Coolant System	203	No	Removed.
Pressure Control and Relief System	204	No	Removed.
Charging and Volume Control System	205	No	Removed.
Chemical Shutdown System	206	No	Removed.
Purification System	207	No	Removed.
Component Cooling System	208	No	Removed.
Primary Plant Corrosion Control System	209	No	Removed.
Primary Plant Sample System	210	No	Removed.
Waste Disposal System	211	Yes	Original system removed, replaced with Temporary Liquid Waste System. [Note]
Shutdown Cooling System	212	No	Removed.
Primary Plant Vent and Drain System	213	No	Removed.
Emergency Core Cooling System	214	No	Removed.
Radiation Monitoring System	215	Yes	Partially removed, in service. [Note]
VC Ventilation and Purge System	216	No	Partially removed, in service.
VC Heating and Cooling System	217	No	Removed.
Post-Accident Hydrogen Control System	218	No	Removed.
Containment Isolation System	219	No	Removed.
Fuel Handling Equipment System	220	Yes	Partially removed, in service. [Note]
SFP Cooling and Purification System	221	Yes	Modified for SFP island. [Note]
Main Steam System	222	No	Removed.
Feedwater System	223	No	Removed.
Steam Generator Blowdown System	224	No	Removed
Emergency Feedwater System	225	No	Removed.
Service Water System	226	Yes	Partially removed; Auxiliary Service Water System installed for SFP island. [Note]
Demineralized Water System	227	Yes	Partially removed, in service. [Note]
Compressed Air System	228	Yes	Original system removed, temporary system in service for SFP. [Note]
Electrical System	229	Yes	Partially removed, in service. [Note]
Heating System	230	No	Partially removed.
Ventilation System	231	Yes	Partially removed, in service. [Note]

TABLE–1: STATUS OF SYSTEMS, STRUCTURES, AND COMPONENTS DESCRIBED IN FSAR

(continued)

System, Structure, or Component	FSAR Section	Needed during Wet Fuel Storage	Status
Fire Protection and Detection System	232	Yes	Partially removed, in service. [Note]
Primary Pump Seal Water System	233	No	Removed.
Safe Shutdown System	234	No	Removed.
Water Cleanup System	235	No	Removed.
Vapor Container	236	No	Placed in lay-up condition.
Reactor Support Structure	237	No	Placed in lay-up condition.
Vapor Container Polar Crane	238	No	Placed in lay-up condition.
Radiation Shielding	249	No	Partially removed/decontaminated.
Neutron Shield Tank	240	No	Removed.
Pipe Chases	241	No	Placed in lay-up condition.
Fuel Transfer Chute	242	No	Partially removed/decontaminated.
Yard Area Crane and Support Structure	243	Yes	In service. [Note]
Ion Exchange Pit	244	Yes	Partial decontamination in 1997, full decon after fuel removed from SFP. North wall required structurally for SFP. [Note]
Primary Vent Stack	245	Yes	In service. [Note]
Spent Fuel Pit and Spent Fuel Pit Building	246	Yes	In service. [Note]
New Fuel Vault	247	Yes	To be decontaminated after fuel removed from SFP. West wall required structurally for SFP. [Note]
Primary Auxiliary Building	248	No	Partially decontaminated.
Diesel Generator Building	249	No	Building demolished.
Waste Disposal Building	250	No	Partially decontaminated.
Safe Shutdown System Building	251	No	Partially decontaminated.
Potentially Contaminated Area Storage Buildings 1 and 2 and Warehouse	252	Yes	PCA Bldgs. No. 1 and 2 to be decontaminated. Warehouse may be used during wet or dry fuel storage.
Compactor Building	253	No	To be decontaminated.
Service Building	254	Yes	Portions of building within the Radiation Control Area are in service. [Note]

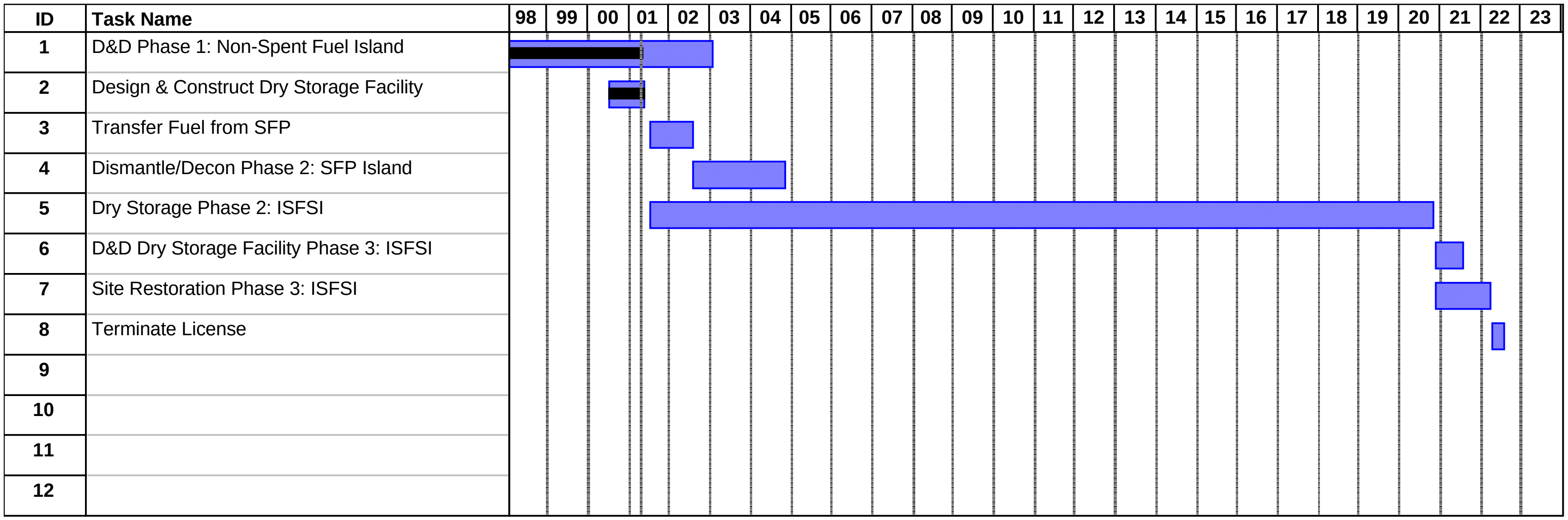
TABLE–1: STATUS OF SYSTEMS, STRUCTURES, AND COMPONENTS DESCRIBED IN FSAR

(continued)

System, Structure, or Component	FSAR Section	Needed during Wet Fuel Storage	Status
Miscellaneous Tanks	255	Yes	Most removed; one tank to remain for storage of demineralized water for SFP. [Note]
Meteorological Tower	256	Yes	In service. Function will be retained during wet or dry storage.

Note: After all spent fuel and Greater-Than-Class-C waste has been removed from the Spent Fuel Pool, the remaining portions of this system or structure that are in service to support SFP operations will be decontaminated and dismantled.

FIGURE 1-1: YNPS DECOMMISSIONING SUMMARY SCHEDULE



100 SPENT FUEL MANAGEMENT PLAN

100.1 Background

This section documents the YNPS spent fuel management plan in accordance with 10CFR50.54(bb) (Reference 100-1). This regulation also requires a funding plan for spent fuel storage. The spent fuel storage cost estimate and funding plan are presented in the PSDAR.

Currently, there are 533 fuel assemblies stored in double tier racks in the Spent Fuel Pit. These fuel assemblies were discharged from the reactor between 1972 and 1992. There also are a number of failed fuel pins that must be consolidated before they are moved from the Spent Fuel Pit. The Spent Fuel Pit also contains greater than Class C (GTCC) low level radioactive waste in the form of canisters containing reactor vessel internals (core baffle). The canisters have the same external dimensions as fuel assemblies. Several miscellaneous low level radioactive items also are stored in the Spent Fuel Pit (e.g. neutron sources, filter cartridges, material from reconstitution activities).

YAEC is currently seeking accelerated acceptance of YNPS's spent fuel by the Department of Energy in accordance with the current fuel disposal contract. The Department of Energy's current position is that they have not yet determined whether priority will be accorded shutdown reactors, or if priority is granted, under what specific circumstances it might be granted. For planning purposes, fuel shipments are assumed to be completed in 2020. This projection is based on the Department of Energy's Acceptance Priority Ranking, Annual Capacity Report, and an extrapolation beyond the 10 year Department of Energy outlook. For planning purposes, YAEC's current decommissioning cost estimate assumes storage of fuel in the Spent Fuel Pit into 2002, at which time it will be transferred to an on-site dry storage facility. Spent fuel is assumed in the PSDAR to remain in the dry storage facility until 2020.

100.2 Spent Fuel Management Strategy

A task force was formed, following the decision to permanently cease power operations, to develop a spent fuel management strategy. The task force completed a comprehensive evaluation of the following spent fuel storage and disposal alternatives (Reference 100-2):

- Continued operation of the Spent Fuel Pit and required auxiliary systems.
- Construction of an independent wet spent fuel storage facility.
- Construction of a dry cask spent fuel storage facility.
- Shipment of the spent fuel off-site.

Based on the task force review, the following spent fuel management strategy was implemented:

- Continue operation of the Spent Fuel Pit and implement any safe, but economically attractive enhancements.

- Urge the Department of Energy to accelerate acceptance of spent fuel or to accept financial responsibility for on-site spent fuel storage.
- Continue evaluations of wet and dry storage options to reflect YAEC and industry developments.
- Initiate preliminary design activities for a dry storage facility.

The decision to initiate preliminary design activities for a dry storage facility is based on the reduced operating costs of the dry storage facility compared to those of the Spent Fuel Pit. The wet storage option also requires operation of support systems, restricting dismantlement activities. Spent Fuel Pit operations will continue until fuel is permanently removed to either an on-site or off-site facility. Section 100.3 presents limitations on Spent Fuel Pit operations during decommissioning activities.

YAEC has concluded that a custom-designed transport and storage cask system is the most viable option. YAEC has contracted with NAC International to transfer the fuel from the Spent Fuel Pit to a dry storage facility. The location for the new storage facility is at the south end of the YNPS site (see Figure 257-1). The 533 fuel assemblies will be transferred into 15 canisters that will contain a maximum of 36 fuel assemblies. The GTCC waste material stored in the Spent Fuel Pit will be placed into a separate canister. The canisters will be placed into a concrete cask which will provide cooling and shielding. The concrete casks will then be placed on the concrete pad located at the south end of the site.

An additional benefit to construction and operation of the dry cask facility is that it will permit decommissioning of the Spent Fuel Pit structure coincident with other dismantlement activities, leaving removal of the relatively simple and essentially uncontaminated dry cask facility when fuel is removed by the Department of Energy.

100.3 Spent Fuel Pit Operation Limitations

YAEC evaluated the option to store fuel in the Spent Fuel Pit during a portion of the dismantlement phase of decommissioning (Reference 100-3). The purpose of the evaluation was to identify safety considerations and limitations on decommissioning activities associated with operating the Spent Fuel Pit concurrent with dismantlement activities. Operation of the Spent Fuel Pit during the dismantlement phase would allow YAEC additional time to pursue early transfer of spent fuel to the Department of Energy without incurring a significant investment associated with the dry cask facility. This option also allowed additional time for the development of a canister system that is compatible with YNPS limitations.

Dismantlement activities are limited when spent fuel is stored in the Spent Fuel Pit. Activities cannot be pursued that could result in the loss of Spent Fuel Pit integrity or in physical damage to the fuel that would reduce subcriticality margin or cause a loss of a coolable geometry. These events can be precluded by incorporating the following provisions in the plan:

- Delay dismantlement of the Ion Exchange Pit until after the fuel is permanently removed from the Spent Fuel Pit.

- The Yard Area Crane has been upgraded to a single-failure-proof crane meeting the requirements of NUREG-0554 with the implementation of EDC 96-305, “Yard Area Crane Technical Specification 3.2 currently limits cask usage in and over the Spent Fuel Pit to a shipping or transfer cask weighing no more than 80 tons.

- Isolate the Fuel Transfer Chute from the Spent Fuel Pit by: (a) resupporting the Fuel Transfer Chute/Spent Fuel Pit penetration assembly to the Spent Fuel Pit using the latch mechanism, (b) filling the annular space between the Fuel Transfer Chute pipe and the Spent Fuel Pit penetration pipe with grout, (c) removing one section of Fuel Transfer Chute pipe uphill of the Lower Lock Valve (LLV), (d) installing a blind flange cap on the LLV, (e) erecting permanent form work and placing a concrete barrier in the LLV pit and (f) installing metal plates above and below the LLV pit to preclude personnel access to this area. This has been completed with the implementation of EDC 95-303, “Fuel Transfer Chute Isolation.”

- Ensure that detailed work planning excludes activities that could result in a drop of a heavy load onto or into the Spent Fuel Pit or Fuel Transfer Chute. Ensure that partial dismantling of equipment and structures does not result in a configuration that could result in failure during an external event (e.g., seismic event) and subsequent collapse onto or into the Spent Fuel Pit or Fuel Transfer Chute. Alternatively, consider modifications to protect the Spent Fuel Pit and the Fuel Transfer Chute from heavy load drops. Technical Specification 3.2 limits movement of loads over the Spent Fuel Pit to those less than 900 pounds, other than for the specific exceptions that have been identified.

- Ensure that demolition explosives that could affect the Spent Fuel Pit structure are not permitted for use until either fuel is permanently removed from the Spent Fuel Pit or an analysis of the impact of explosives on the Spent Fuel Pit structure is completed.

Movement of fuel from the Spent Fuel Pit to the on-site dry cask fuel storage facility will require a separate evaluation to ensure that there are no unacceptable interactions between fuel movement and decommissioning activities.

Although it is important to take actions to prevent a loss of spent fuel cooling capability, the consequences of such an event are not severe. As of January 1994, greater than four weeks must elapse without re-establishing cooling or adding make-up water before the water remaining in the Spent Fuel Pit is insufficient to provide shielding adequate for operator response in the Spent Fuel Pit Building. At that time a make-up water flow rate of about 1 gpm is needed to replace water lost through evaporation. Several diverse sources of make-up are available including demineralized water, fire water, auxiliary service water, or Sherman Reservoir water. Water may be injected to the Spent Fuel Pit through installed or portable pumps as well as gravity feed. Adequate time is available either to re-establish Spent Fuel Pit cooling or to provide make-up water to maintain Spent Fuel Pit inventory.

Although scenarios that result in a loss of spent fuel cooling capability can be mitigated without any significant consequence, the capability should be protected during decommissioning. Dismantlement activities near and around the Spent Fuel Pit Cooling System and other support systems should be controlled to prevent damage to these systems. This may be accomplished by

physically protecting the systems or by establishing safe load paths and protective zones around the systems.

In addition to maintaining the Technical Specification required water level over the fuel, programs are in place to ensure that water chemistry is adequately maintained to prevent spent fuel degradation. Purification and filtration are accomplished by using self-contained, submersible ion-exchange and filter units. These processes are independent of the operation of the Spent Fuel Pit Cooling System. Periodic sampling in accordance with an administrative plant procedure ensures that water chemistry is maintained within plant administrative limits for acidity, conductivity, chlorides, fluorides and radionuclides.

YAEC will maintain the systems and components required to support spent fuel storage in accordance with the possession only license and other administrative and implementing procedures. The maintenance program consists of corrective maintenance, preventive maintenance, and surveillances. YAEC uses a Predictive Maintenance Program to assure reliable operations of the systems and components that support spent fuel cooling. In addition, the operations staff provides daily monitoring of the performance of the systems and components that support spent fuel storage.

100.4 Special Nuclear Material Control

Information regarding Special Nuclear Material Control is available for inspection by authorized personnel.

REFERENCES

100-1 59-FR-10267, Notification of Spent Fuel Management and Funding Plans By Licensees of Prematurely Shut Down Power Reactors, March 4, 1994.

100-2 YRP 435/92, Spent Nuclear Fuel Storage Study Report and Recommendations, B. W. Holmgren, J. M. Buchheit, R. A. Mellor to J. K. Thayer, October 9, 1992.

100-3 YRP 303/93, Impact of Wet Spent Fuel Storage on Decommissioning, P. A. Rainey to R. A. Mellor, July 15, 1993.

200 DECOMMISSIONING ACTIVITIES AND PLANNING

200.1 Introduction

This section presents a description of decommissioning activities and tasks for the implementation of the dismantlement activities at YNPS. The information presented in this section reflects initial planning of decommissioning activities. YAEC intends to complete more detailed planning prior to initiating each decommissioning activity. Detailed planning includes engineering design; ALARA planning; and cost, schedule, and resource refinement.

Decommissioning will be completed in three phases. The first phase consists of the decontamination and dismantlement of certain structures systems and components which do not support fuel storage. After the removal of all spent fuel from the Spent Fuel Pit (SFP), the second phase of decommissioning will involve the dismantlement and decontamination of the SFP and the remaining systems, structures and components. The final phase of decommissioning will involve the termination of the possession only license. License termination will occur after all spent fuel has been removed off-site and, if dry cask storage facility is constructed, decommissioning of that facility. All decommissioning activities will be accomplished with no significant adverse environmental impacts.

200.2 Decommissioning Plan Approval

A Decommissioning Plan (Reference 500-1) was prepared and submitted in response to the requirements of 10CFR50.82, “Application for Termination of License,” using guidance from Draft Regulatory Guide DG-1005 (Reference 500-2). The Decommissioning Plan provided the framework for the decommissioning process by which the YNPS site will be returned to an unrestricted use condition.

The Decommissioning Plan was approved by NRC on February 14, 1995 (Reference 500-3). Following NRC approval, the Decommissioning Plan was incorporated into the FSAR in accordance with Reference 500-4 and the FSAR was conformed to the Decommissioning Plan using the process described in 10CFR50.71(e). Essential features and functions that will be relied upon during decommissioning are now described in the FSAR.

In 2000, Yankee created a Post-Shutdown Decommissioning Activities Report (PSDAR) within the FSAR. NRC Draft Regulatory Guide DG-1071 recommends that licensees with an approved Decommissioning Plan (Dplan) “extract pertinent detail from the decommissioning plan and submit a PSDAR update in the format and content specified by [DG-1071].” Based on the NRC draft guidance, Yankee segregated, updated and condensed certain information concerning post-shutdown decommissioning activities in a manner that conforms to the standard format and content of a PSDAR.

200.3 Decontamination and Dismantlement: General Information

200.3.1 Overview

This section presents a general description of the decontamination and dismantlement activities that are necessary to decommission YNPS. The information presented in this section provides sufficient detail to address the adequacy of the programs and procedures, ensuring safe and economic decommissioning of YNPS. The information in this section has been incorporated into the more detailed planning that will be completed prior to initiating each decommissioning activity. Detailed planning includes engineering design; ALARA planning; and cost, schedule, and resource refinement. Sections 201 through 256 present specific information for YNPS systems, structures and components.

The description of decontamination and dismantlement options includes the words “should” and “must” to describe alternatives. The word “should” implies that the alternative is preferred, however, alternatives may be available that are equally acceptable. The word “must” implies that the alternative is based on a programmatic, regulatory or safety analysis requirement. If another alternative is chosen, the corresponding requirements must be re-evaluated to ensure that the original intent is not adversely affected.

Before the start of the dismantlement activities, a decommissioning administrative and engineering organization will be mobilized. The following activities will occur:

- Initiation of detailed project planning.
- Preparation of engineering specifications and procedures.
- Procurement of special equipment needed to support decommissioning.
- Negotiation of service contracts required for decommissioning activities.
- Reactivation and return to service of systems required for decommissioning.

The engineering and preparation phase is followed by the initiation of plant dismantlement activities. The contaminated systems will be removed, packaged, and either shipped to an off-site processing facility or shipped directly to a low level radioactive waste disposal facility. Decontamination of plant structures will be completed concurrently with the equipment and system removal process. Structure decontamination will include a variety of techniques ranging from high pressure water washing to removal of concrete to allow release of the structures. Contaminated structural material will be packaged and either shipped to a processing facility for decontamination or shipped directly to a low level radioactive waste disposal facility.

Following the removal of contaminated systems, structures and components, a comprehensive final radiation survey will be conducted. The survey will verify that radioactivity has been reduced to sufficiently low levels allowing unrestricted release of the site. Successful completion of the final survey will be demonstrated through a verification survey completed by an independent contractor selected by the NRC.

200.3.2 Detailed Planning and Engineering Activities

YAEC is the prime entity with license responsibility for YNPS decommissioning. In this position, YAEC has direct control and oversight over all decommissioning activities. This role is similar to that taken by YAEC during the 31 year operation of YNPS. In that role, YAEC provided operational, technical, licensing, project management, and contractor oversight for all plant systems, structures and components.

Detailed project implementation plans will be developed to support decontamination and dismantlement activities before these activities are initiated. The plans will be used as a project management tool to support detailed engineering activities and ALARA Program implementation, estimate decommissioning labor requirements, and manage decommissioning cost and schedule.

Decommissioning work packages will be completed for all decommissioning activities. The work packages will be developed using an administrative plant procedure and similar controls to those used to support YNPS operations:

- <u>Engineering Specifications</u> - Engineering specifications will be developed for all significant system, structure, and component removal activities. A procedure will be used to provide guidance for the preparation of engineering specifications for decommissioning activities. All significant decommissioning activities receive appropriate safety and technical reviews. Engineering specifications will receive an Independent Safety Review (ISR), or equivalent, prior to implementation.

- <u>Decommissioning Procedures</u> - All decommissioning activities will be completed using procedures which have received an ISR or equivalent prior to implementation. The procedures will be developed and controlled in accordance with an administrative plant procedure. This procedure presents administrative controls for the format, content, review, and approval of all procedures used at the plant. Applicable elements of the Radiation Protection Program and Occupational Safety Program will be integrated into the procedures (e.g., ALARA Program, Radioactive Waste Minimization Program).

200.3.3 General Decontamination and Dismantlement Considerations

The following are general decontamination and dismantlement considerations that will be incorporated into the decommissioning work packages for the systems, components and structures at YNPS. Specific decontamination and dismantlement considerations are presented in Sections 201 through 256 for applicable systems, structures and components.

a. Caution must be used when working in areas which contain systems or structures that support spent fuel cooling and storage (e.g., Spent Fuel Pit Building, Primary Auxiliary Building Fan Room and Screenwell House). The systems and structures cannot be affected by removal activities. Work packages must include specific steps either to physically protect the systems and structures or to establish safe load paths and protective zones around the systems and structures (Section 100.3).

b. Dismantlement activity planning must consider the impact of seismic events on components that are affected by removal activities. These components should be evaluated and physically supported, as appropriate, to limit the off-site dose resulting from a release of radioactivity to within the accident analysis limits (Section 400). The impact of seismic events must also be evaluated during planning of dismantlement activities in proximity of the Spent Fuel Pit. The purpose of this evaluation is to ensure that partial dismantling of equipment and structures does not result in a configuration that could fail during a seismic event, subsequently collapsing onto or into the Spent Fuel Pit or Fuel Transfer Chute.

c. Hazardous materials and wastes must be processed in accordance with the YNPS Hazardous Waste Management Program (Section 511), including the following:

- Asbestos containing material (e.g., insulation) must be removed and processed in accordance with the YNPS Asbestos Control Program (Section 511.4.6). This activity should be scheduled prior to initiating equipment removal in an area.

- Structures and components containing lead based coatings must be processed in accordance with the guidance presented in Section 511.4.7.

- Systems and components which contained or were immersed in chromated solutions must be free of chromate residue before shipment off-site. The systems and components must be rinsed, if necessary, to ensure that loose surface chromate residue is removed.

- Paint coatings containing solid Polychlorinated Biphenyls (PCBs) must be managed in accordance with all federal and state regulations.

d. The capability to isolate the Vapor Container from the environment and mitigate the consequences of a significant radioactive release, shall be maintained as long as practicable during decontamination and dismantlement activities in the Vapor Container. Vapor Container isolation is the closure of penetrations and openings to restrict transport of airborne radioactivity from the Vapor Container atmosphere to the environment. Pressure retention capability is not necessary.

   If this capability cannot be established, activities involving significantly contaminated or activated components must be suspended. This consideration should not preclude the removal of penetrations and attachments to the Vapor Container shell, provided that the opening is closed in a timely manner.

e. Decommissioning activities that use liquids must ensure that contaminated liquids will be processed by the liquid waste processing system. Additionally, existing or supplemental barriers must be used to ensure that inadvertent spills from these activities are contained within the liquid waste processing system.

f. The following considerations must be incorporated into tank and vessel sludge removal activities:

- The method used must ensure that any liquid inadvertently discharged from the system is contained in the plant liquid waste processing system.

- Sludge removed from the system must be stabilized prior to shipment.

- Waste water must be processed and analyzed before discharging from the facility.

- The use of a high pressure water rinse should be used before dismantlement, if necessary, to reduce internal contamination levels.

g. Radioactive particulate emissions must be filtered and monitored to the maximum extent practicable. The following must be implemented:

- The VC Ventilation and Purge System must be maintained in operation during decontamination and dismantlement activities in the Vapor Container.

- The Ventilation System must be maintained in operation during decontamination and dismantlement activities in the Primary Auxiliary Building cubicle area, Waste Disposal Building, Spent Fuel Pit Building, and the Fuel Transfer Enclosure.

- Local HEPA filtration systems must be used when activities could result in the release of significant radioactive particulates or hazardous materials. The local HEPA filtration systems should exhaust to areas served by the Ventilation System when used outside of the Vapor Container to monitor particulate releases. If the work area is a significant distance from the plant Ventilation System, it may not be practical to meet this requirement. In these cases, monitoring of the HEPA Filtration System exhaust will be performed. Prior to initial deployment and periodically, each HEPA System is tested against Dioctyl Phthalate (DOP) with an acceptance criterion of 99.95% filter retention. This ensures that airborne particulate releases to the environment have been reduced to as low as is reasonably achievable.

h. Electrical and pneumatic services must be isolated from the systems, components and structures prior to dismantlement.

The following should be implemented:

- Pumps, fans, heaters, motor operated valves, motor operated dampers and instrumentation power sources should be isolated and disconnected from station electrical and control systems at the motor control centers, supply breakers, fuse blocks, and at the equipment.

- Pneumatically operated components and instrumentation should be isolated from the Compressed Air System at the root and equipment isolation valves.

i. Openings in components must be enclosed with a protective cover to confine internal contamination.

j. Explosive methods must not be used during the YNPS decommissioning to remove contamination.

k. Before removing contaminated systems, structures and components with significant external contamination, they should be wiped down to remove external contamination or painted with a coating to stabilize external contamination.

l. Contaminated piping and tubing should be dismantled as follows:

- Large bore piping (greater than 2-1/2 inch diameter) should be cut into manageable lengths. Significantly contaminated systems must be cut using mechanical methods to minimize the generation of airborne contamination.

- Small bore piping (less than 2-1/2 inch diameter) and tubing should be cut using an appropriate method based on the radiological conditions.

- Remote cutting systems should be used as appropriate to maintain worker exposure as low as is reasonably achievable.

- Open pipe ends must be enclosed with a protective cover to confine contamination to the inside of the pipe.

- Piping penetrations should be cut as close as practicable to the Vapor Container shell. The opening in the Vapor Container shell should be covered once the piping is removed.

- Underground piping must be visually examined after it is excavated to ensure that it is physically sound prior to cutting and removal.

m. Contaminated supports should be removed in conjunction with equipment removal activities.

n. Systems and components should be removed from areas and buildings prior to the commencement of area structural decontamination activities. The block shield walls in the Primary Auxiliary and Waste Disposal Buildings have been removed as necessary to permit removal of systems and components.

o. Embedded contaminated piping, conduit, ducts, plates, channels, anchors, sumps and sleeves should be removed or decontaminated during area and building structural decontamination or demolition activities.

p. Centralized processing and cutting stations should be considered to facilitate packaging of components for shipment to an off-site processing facility or a low level radioactive waste disposal facility.

200.3.4 Decontamination and Dismantlement Process

The decontamination and dismantlement of contaminated systems, structures and components may be accomplished by either decontamination in place, removal and decontamination, or removal and

disposal. A combination of these methods may be utilized to reduce contamination levels and reduce worker radiation exposures.

In general, contaminated and potentially contaminated systems, structures and components will be removed as follows:

- Radiological characterization survey data will be used to identify the systems, structures and components to be decontaminated and dismantled. The type of contamination associated with the systems and components is presented in Table 200.1.

- Decommissioning work packages with sufficient detail will be developed, reviewed and approved in accordance with project and plant programs and procedures (Section 200.3.2)

- Plant tag-out procedures will be used to de-energize electrical and control equipment, isolate and drain fluid systems, and isolate and depressurize pneumatic systems. Radiation Protection procedures will be used to ensure radiological requirements for control of contamination, worker protection and ALARA program are satisfied. Occupational Safety standards will be observed.

- All components will be identified prior to removal. The components will then be removed using the techniques and methods as specified in the decommissioning work packages. All openings in components will be covered and sealed to minimize the spread of contamination. The components may be moved to a processing area for volume reduction and packaging into containers for shipment to a processing facility for decontamination or low level radioactive waste disposal facility.

- Contaminated concrete and structural steel components will be decontaminated or removed when all contaminated and uncontaminated systems and equipment have been removed from the area or building. The contaminated concrete will then be removed and packaged into containers for shipment to a low level radioactive waste disposal facility. Contaminated structural steel components may be moved to a processing area for volume reduction and packaging into containers for shipment to a processing facility for decontamination or low level radioactive waste disposal facility.

- Buried contaminated components (e.g., piping, drains, conduit) will be excavated. After excavation, the components will be examined to ensure that they are physically sound prior to cutting and removal. Most buried contaminated piping is located in steel conduits (i.e., pipes enclosed in pipes). Contamination controls will be modified as necessary if the components are significantly degraded.

- A final termination survey will be performed to verify removal of contamination to below release levels.

A summary of currently available methods for decontamination of plant equipment and structures is presented in Table 200.2. The methods presented in this table are the most practicable and widely utilized at the time that the Decommissioning Plan was generated. However, new decontamination technologies will be adopted, if appropriate.

A summary of currently available methods for cutting plant equipment and structures is presented in Tables 200.3 and 200.4. The methods presented in this table are the most practicable and widely utilized at the time that the Decommissioning Plan was generated. However, new dismantlement technologies will be adopted, if appropriate.

200.4 Materials Cutting Station

A centralized processing and cutting station may be implemented to facilitate packaging of components for shipment to an off-site processing facility or a low level radioactive waste disposal facility. The processing and cutting station atmospheric control must be done in accordance with the requirements of Section 508.3.

200.5 Decontamination and Dismantlement Plan: Systems, Structures and Components

200.5.1 Overview

As required, Sections 201 through 256 present a specific description of the decontamination and dismantlement activities that are necessary to decommission each potentially contaminated system, structure and component at YNPS. The information presented in these sections reflect the initial planning of decommissioning activities and will be incorporated into the more detailed planning that will be completed prior to initiating any decommissioning activity. Detailed planning includes engineering design; ALARA planning; and cost, schedule, and resource refinement.

The following systems were not described in the Decommissioning Plan and are not discussed in the FSAR:

Auxiliary Steam System (excluding the Auxiliary Boilers)
Condensate System
Circulating Water System
Turbine Oil System
Generator Hydrogen and Seal Oil System

These systems have no significant interaction with the decommissioning of YNPS. However, the systems may be dismantled during the dismantlement periods.

TABLE 200.1

System Contamination

Section	System	Internal	External	Extent of Contamination
203	MAIN COOLANT SYSTEM	X	X	ENTIRE SYSTEM
204	PRESSURE CONTROL AND RELIEF SYSTEM	X	X	ENTIRE SYSTEM
205	CHARGING AND VOLUME CONTROL SYSTEM	X	X	ENTIRE SYSTEM
206	CHEMICAL SHUTDOWN SYSTEM	X	X	ENTIRE SYSTEM
207	PURIFICATION SYSTEM	X	X	ENTIRE SYSTEM
208	COMPONENT COOLING WATER SYSTEM	X	X	ENTIRE SYSTEM
209	PRIMARY PLANT CORROSION CONTROL SYSTEM		X	ENTIRE SYSTEM
210	PRIMARY PLANT SAMPLE SYSTEM	X	X	ENTIRE SYSTEM
211	WASTE DISPOSAL SYSTEM	X	X	ENTIRE SYSTEM
212	SHUTDOWN COOLING SYSTEM	X	X	ENTIRE SYSTEM
213	PRIMARY PLANT VENT AND DRAIN SYSTEM	X	X	ENTIRE SYSTEM
214	EMERGENCY CORE COOLING SYSTEM	X	X	ENTIRE SYSTEM
215	RADIATION MONITORING SYSTEM		X	ENTIRE SYSTEM
216	VC VENTILATION AND PURGE SYSTEM	X	X	ENTIRE SYSTEM
217	VC HEATING AND COOLING SYSTEM	X	X	ENTIRE SYSTEM
218	POST-ACCIDENT HYDROGEN CONTROL SYSTEM	X	X	ENTIRE SYSTEM
219	CONTAINMENT ISOLATION SYSTEM		X	ENTIRE SYSTEM
220	FUEL HANDLING EQUIPMENT SYSTEM	X	X	ENTIRE SYSTEM
221	SFP COOLING AND PURIFICATION SYSTEM	X	X	ENTIRE SYSTEM
222	MAIN STEAM SYSTEM		X	PARTIAL SYSTEM
223	FEEDWATER SYSTEM		X	PARTIAL SYSTEM
224	STEAM GENERATOR BLOWDOWN SYSTEM	X	X	PARTIAL SYSTEM
225	EMERGENCY FEEDWATER SYSTEM		X	PARTIAL SYSTEM
226	SERVICE WATER SYSTEM		X	PARTIAL SYSTEM
227	DEMINERALIZED WATER SYSTEM		X	PARTIAL SYSTEM
228	COMPRESSED AIR SYSTEM		X	PARTIAL SYSTEM
229	ELECTRICAL SYSTEM		X	PARTIAL SYSTEM
230	HEATING SYSTEM	X	X	PARTIAL SYSTEM
231	VENTILATION SYSTEM	X	X	ENTIRE SYSTEM
232	FIRE PROTECTION AND DETECTION SYSTEM		X	PARTIAL SYSTEM
233	PRIMARY PUMP SEAL WATER SYSTEM		X	ENTIRE SYSTEM
234	SAFE SHUTDOWN SYSTEM	X	X	PARTIAL SYSTEM
235	WATER CLEANUP SYSTEM		X	PARTIAL SYSTEM

TABLE 200.2

Decontamination Methods

Method	Advantages	Disadvantages
Carbon Dioxide Blasting	• Low waste volume	• High operating costs • HEPA filtration required to control airborne contamination • High setup costs
Abrasive Blasting	• Very effective for surface contamination	• Large waste volumes • HEPA filtration required to control airborne contamination
Hydro Blasting	• Remote operation possible • Very effective for surface contamination • Easy to use	• Large waste volumes • HEPA filtration required to control airborne contamination
Strippable coatings	• Easy to use • Good for fixing loose surface contamination	• Only effective on loose contamination
Scarifying	• Effective on coated and uncoated surfaces • Removes concrete to ¼ inch deep	• Cannot cut rebar • Vacuum with HEPA must be used to control waste and dust
Scabbling	• Removes concrete to ¼ inch deep per pass • Easy to control removal depth per pass	• Cannot cut rebar • Vacuum with HEPA must be used to control waste and dust • Not as effective on coated surfaces
Spalling	• Low airborne contamination • Good for limited access and small areas	• Slow process • Cannot cut rebar
Vacuum Cleaning	• Easy to use • Fast removal times	• Only effective on loose contamination • HEPA filtration required to control airborne contamination

TABLE 200.3

Mechanical Cutting/Removal Methods

Method	Applicability	Advantages	Disadvantages
Machining	• Contaminated pipe cutting • Vessel segmentation	• Minimum airborne contaminatation • Easy setup • Remote operate • Quick cutting times	• Unit is heavy in larger sizes
Abrasive Water-Jet	• Tank segmentation • Concrete cutting	• Remote operation possible	• Large amounts of liquid waste • HEPA filtration required to control airborne contamination • Slow cutting times
Abrasive Wheel	• Steel cutting	• Inexpensive • Easy to setup and operate • Can cut rebar and imbedded steel	• HEPA filtration required to control airborne contamination • Slow cutting times
Diamond Wire	• Concrete cutting	• Capable of cutting thick concrete • Can cut rebar and imbedded steel • Remote operation possible	• Slow cutting times • Large amounts of liquid waste
Metal Disintegration Machining (MDM)	• Vessel segmentation	• Capable of cutting thick steel • High precision obtainable • Remote operation possible	• Slow cutting times • Complex control system required
Mechanical Shears	• Tubing, cable and pipe cutting	• Quick cutting times • Best for tubing and sheet metal	• Cannot be used on large times
Band and Reciprocating Saws	• Contaminated pipe and steel cutting	• Easy to use	• Slow cutting times • Cannot be used on large items
Impact Hammer	• Concrete removal	• Inexpensive • Fast removal rates	• HEPA filtration required to control airborne contamination • Creates rubble

TABLE 200.4

Thermal Cutting/Removal Methods

Method	Applicability	Advantages	Disadvantages
Plasma Arc	• Tank segmentation • Uncontaminated pipe cutting • Structural steel cutting • Contaminated piping (with appropriate radiological controls)	• Fast cutting times • Remote operation possible	• HEPA filtration required to control airborne contamination from contaminated component removal • Large power supply required
Oxy-Fuel	• Tank segmentation • Uncontaminated pipe cutting • Structural steel cutting • Contaminated piping (with appropriate radiological controls)	• Easy to use • Fast cutting times • Remove operation possible	• HEPA filtration required to control airborne contamination from contaminated component removal • Ineffective on stainless steel • Uses inflammable gases
Electric Discharge Machining (EDM)	• Vessel segmentation	• Capable of cutting thick steel • High precision obtainable • Remote operate possible	• Slow cutting times • Complex control system required
Flame Cutting	• Concrete cutting	• Capable of cutting thick concrete structures • Low waste volumes	• HEPA filtration required to control airborne contamination • Difficult to use
Arc Saw	• None at this time	• Capable of cutting thick steel • Remote operation possible	• HEPA filtration required to control airborne contamination • Large power supply required • Heavy weight of unit • Limited experience with unit

201 REACTOR VESSEL

201.1 Status

The Reactor Vessel has been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

202 STEAM GENERATORS

202.1 Status

The Steam Generators have been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

203 MAIN COOLANT SYSTEM

203.1 Status

The Main Coolant System has been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

204 PRESSURE CONTROL AND RELIEF SYSTEM

204.1 Status

The Pressure Control and Relief System has been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

205 CHARGING AND VOLUME CONTROL SYSTEM

205.1 Status

The Charging and Volume Control System has been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

206 CHEMICAL SHUTDOWN SYSTEM

206.1 Status

The Chemical Shutdown System has been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

207 PURIFICATION SYSTEM

207.1 Status

The Purification System has been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

208 COMPONENT COOLING WATER SYSTEM

208.1 Status

The Component Cooling Water System has been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

209 PRIMARY PLANT CORROSION CONTROL SYSTEM

209.1 Status

The Primary Plant Corrosion Control System has been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

210 PRIMARY PLANT SAMPLE SYSTEM

210.1 Status

The Primary Plant Sample System has been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

211 TEMPORARY WASTE WATER PROCESSING ISLAND SYSTEM

211.1 Description

The Temporary Waste Water Processing Island System receives, contains, treats and safely disposes all liquid radioactive wastes. Waste water generated as a result of decommissioning activities is routed to a 20,000 gallon waste water storage tank (TK-81). The tank currently accepts water only from the radioactive lab sump discharge line; however, piping is installed to accommodate future connections from the plant as needed. The waste water is pumped to the 20,000 gallon storage tank and is then transferred to the evaporator equipment enclosure for processing.

Housed in the evaporator equipment enclosure are three pumps (feed, distillate and bottoms), heat exchanger with pump for the 20,000 gallon tank heating system, evaporator, High Integrity Container (HIC), filter, and two ion exchange polishers (Drawing Number 9699-FM-103A). By batch, waste water feed is transferred from the storage tank to the evaporator via an air-operated feed pump. The gas-fired evaporator then boils off water, creating steam, that is then condensed by a 60 ton chiller. The distillate is then collected into a 22 gallon tank with a level control to start/stop the air-operated distillate pump. The condensed distillate is further purified by a filter and two ion exchange polishers. Each ion exchanger is installed in series with the ability to bypass either ion exchange column. After purifying the water, the water is transferred to the two 5,000 gallon test tanks for sampling and discharge via the Auxiliary Service Water (ASW) System or, if required, makeup for the Spent Fuel Pit. The existing process radiation monitor on the ASW line will alarm and isolate the effluent stream from being discharged in the event the liquid effluent radiation level exceeds the RM setpoint. A HEPA/fan unit draws air through the equipment enclosure and is routed to the primary vent stack for monitoring and discharge. The bottoms are transferred to the HIC for later processing by an off-site disposal facility.

The heating system uses a freestanding, gas-fired boiler. The unit is designed so that it can be installed without an enclosure. A second heat exchanger and pump skid for the 20,000 gallon tank is located in the equipment enclosure. The package heating boiler and the evaporator are supplied fuel from two 1,000 gallon propane tanks located southeast of the warehouse. The heating system water is treated with antifreeze to eliminate the need for freeze protection during periods of prolonged winter downtime. Additionally, all piping external to the evaporator equipment enclosure that has potentially contaminated water is heat-traced and insulated.

Most of the major equipment (20,000 gallon tank, the equipment enclosure, the two 5,000 gallon test tanks, and the package pool boiler) is located in the area east of the Spent Fuel Pit Building, adjacent to Fire Hose House 15. The propane storage tanks are located southeast of the warehouse.

The Temporary Waste Water Processing Island System will be used during the dismantlement periods and may be used to process water from spent fuel pit operations and decommissioning activities.

211.2 Decontamination and Dismantlement Considerations

General decontamination and dismantlement considerations are presented in Section 200.3. The following are specific system considerations if the system is not reused at a different site:

- Temporary Waste Water Processing Island
    - The Temporary Waste Water Processing Island should be isolated at the connections to the Plant Ventilation, Auxiliary Service Water and Rad Lab Sump Systems.
    - Sludge should be removed from the 20,000 gallon storage tank prior to dismantlement of the system.
    - The evaporator should be dismantled into the largest sections practicable which can be removed and processed for shipment.

212 SHUTDOWN COOLING SYSTEM

212.1 Status

The Shutdown Cooling System has been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

213 PRIMARY PLANT VENT AND DRAIN SYSTEM

213.1 Status

The Primary Plant Vent and Drain System has been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

214 EMERGENCY CORE COOLING SYSTEM

214.1 Status

The Emergency Core Cooling System has been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

215 RADIATION MONITORING SYSTEM

215.1 Description

The Radiation Monitoring System monitors plant radiological conditions through two subsystems:

- Process Radiation Monitoring Subsystem - measures radiation levels of selected systems or processes, providing indication, alarms, and limited control functions.

- Area Radiation Monitoring Subsystem - measures radiation levels in selected areas throughout the plant, providing local and remote indication and alarms.

The system consists of detectors, and associated instrumentation and controls.

The following components of the Process Radiation Monitoring Subsystem are required to support plant operations during the dismantlement period:

- Auxiliary Service Water/Liquid Waste Effluent Channel - This channel monitors Auxiliary Service Water (ASW) liquid used to cool the Spent Fuel Pit (SFP), and also, purified liquid effluent from the Temporary Waste Water Processing Island System before it is discharged to Sherman Reservoir. If any of the following conditions occur, a valve on the liquid effluent discharge line will close, terminating the release and trip the operating SFP cooling pump:

  - A high or failure alarm from the ASW/liquid effluent radiation monitor.

  - Loss of power to the ASW/liquid effluent radiation monitor or control circuit.

  The Off-Site Dose Calculation Manual (ODCM) has provisions for operation of the SFP Cooling System with the ASW radiation monitor out of service, such that an automatic pump trip will not occur. The ODCM also has provisions for a liquid effluent release with the ASW radiation monitor out of service, such that an automatic trip valve closure will not occur. Operation of ASW for SFP cooling and release of liquid effluent will continue throughout decommissioning to support dismantlement and decontamination activities.

- Primary Vent Stack Channel - This channel monitors airborne releases from ventilated areas of the primary side of the plant before release to the environment. Airborne release monitoring will continue throughout decommissioning until all ventilated areas are decontaminated (Drawing Number 9699-FK-13A).

The following Area Radiation Monitoring channels, located in potentially contaminated areas, will be used to monitor conditions during system and component dismantling activities:

- Spent Fuel Pit manipulator crane during component movement activities
- Radiation Control Area Control Point
- Primary Auxiliary Building (fan room)

These channels provide local indication, and also provide indication and alarm in the Control Room.

The equipment will be removed as the systems that they monitor are dismantled. The Area Radiation Monitoring equipment will remain in operation until all contaminated process systems have been removed from the area and will then be removed prior to the commencement of area and building decontamination activities. The detector locations may be changed to facilitate removal activities if the new location provides comparable detection capability.

215.2 Decontamination and Dismantlement Considerations

General decontamination and dismantlement considerations are presented in Section 200.3. The following is a specific system consideration:

- The Radiation Monitoring System in uncontaminated areas of the plant should be removed as part of the site dismantlement and restoration process.

216 VC VENTILATION AND PURGE SYSTEM

216.1 Description

Description of VC Ventilation and Purge System has been incorporated in Section 231, Ventilation System.

216.2 Decontamination and Dismantlement Considerations

- General decontamination and dismantlement considerations are presented in Section 200.3. Specific system considerations have been incorporated in Section 231, Ventilation System.

217 VC HEATING AND COOLING SYSTEM

217.1 Status

The VC Heating and Cooling System has been partially removed or retired in place, and is no longer in service. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

218 POST-ACCIDENT HYDROGEN CONTROL SYSTEM

218.1 Status

The portion of the Post-Accident Hydrogen Control System located within the Vapor Container has been permanently removed. The remaining portion of the system has been retired in place. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

219 CONTAINMENT ISOLATION SYSTEM

219.1 Status

The Containment Isolation System has been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

220 FUEL HANDLING EQUIPMENT SYSTEM

220.1 Description

The Fuel Handling Equipment System supports the handling of fuel and irradiated components by providing the following major functions:

- Underwater handling of irradiated fuel in the Spent Fuel Pit
- Storage of irradiated fuel in the Spent Fuel Pit
- Movement of casks to and from the Spent Fuel Pit

The system consists of the Spent Fuel Pit manipulator crane and yard area crane, fuel inspection equipment, grappling fixtures, fuel storage racks, and the necessary associated controls and instrumentation.

The Spent Fuel Pit Manipulator Crane is a trolley-mounted crane on a bridge that moves along rails set in concrete at the top of the Spent Fuel Pit. A rotating turret is mounted on the trolley that supports a rigid tool boom. The tool boom is fitted with a grappling tool which can engage the top end of a fuel assembly. The grappling tool is latched by spring action and unlatched pneumatically. The tool is designed to not disengage when loaded. The manipulator crane is also fitted with north and south outrigger davits from which electric hoists are hung. Manual handling fixtures are used with the outrigger hoists to handle components (including fuel) within the Spent Fuel Pit.

The Spent Fuel Pit fuel handling equipment will be used intermittently until fuel is permanently removed from the Spent Fuel Pit. While fuel is stored in the Spent Fuel Pit during dismantlement activities, special precautions will be instituted to protect the Spent Fuel Pit and associated support systems from physical damage and other adverse conditions that could occur.

The Fuel Handling Equipment System and components are located in the Spent Fuel Pit Building. This system is required for the removal of fuel and irradiated materials from the Spent Fuel Pit.

221 SFP COOLING AND PURIFICATION SYSTEM

221.1 Description

The SFP Cooling and Purification System cools and purifies Spent Fuel Pit water. The system transfers decay heat from the fuel stored in the Spent Fuel Pit to the Auxiliary Service Water System and controls the concentration of radionuclides in the Spent Fuel Pit water. The system consists of two circulating pumps, one heat exchanger, one ion exchange unit, one filter unit, and the necessary associated valves, piping, fittings and instrumentation (Drawing Number 9699-FM-101).

The system uses one of two circulating pumps to move water between the Spent Fuel Pit and a heat exchanger, maintaining water temperature below the design limit of 150°F. Either cooling pump can take suction from either the north or south end of the Spent Fuel Pit and discharge into the west side of the Spent Fuel Pit. The purification function is accomplished by self-contained ion exchange/filter units installed in the Spent Fuel Pit, independent of the Cooling System.

The SFP Cooling and Purification System is not required to support dismantlement activities. While fuel is stored in the Spent Fuel Pit during dismantlement activities, special precautions will be instituted to protect the system from physical damage and other adverse conditions that could occur.

The SFP Cooling and Purification System and components are located in the Spent Fuel Pit Building. The SFP Cooling and Purification System will remain in service until the fuel is permanently removed from the Spent Fuel Pit.

221.2 Decontamination and Dismantlement Considerations

General decontamination and dismantlement considerations are presented in Section 200.3. The following are specific system considerations:

- The SFP Cooling and Purification System should be isolated at the connections to the Demineralized Water and Auxiliary Service Water Systems.

222 MAIN STEAM SYSTEM

222.1 Status

The Main Steam System has been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

223 FEEDWATER SYSTEM

223.1 Status

The Feedwater System has been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

224 STEAM GENERATOR BLOWDOWN SYSTEM

224.1 Status

The Steam Generator Blowdown System has been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

225 EMERGENCY FEEDWATER SYSTEM

225.1 Status

The Emergency Feedwater System has been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

226 AUXILIARY SERVICE WATER SYSTEM

226.1 Description

The Auxiliary Service Water System (ASWS) supports plant operations by supplying water from Sherman Reservoir for the following major uses:

- Cooling water for the Spent Fuel Pit
- Dilution water for waste water releases

The system consists of one auxiliary service water pump and the necessary associated valves, piping, fittings and instrumentation (Drawing Number 9699-FM-101). The pump, installed in the Screenwell House, circulates Sherman Pond water through the Spent Fuel Pit Cooling System (SFPCS) heat exchanger and discharges it back into Sherman Pond. An effluent radiation monitor is installed downstream of the heat exchanger (see Section 215). As an added measure of leak protection, a positive differential pressure is maintained between the ASWS and the SFPCS at the heat exchanger.

The Auxiliary Service Water System and components are located in the Screenwell House, Spent Fuel Pit Building and Yard Area. The system will be used to support decontamination and dismantlement activities. The Auxiliary Service Water System is required to support Spent Fuel Pit operations.

226.2 Decontamination and Dismantlement Considerations

General decontamination and dismantlement considerations are presented in Section 200.4. The following is a specific system consideration:

- The Auxiliary Service Water System should be isolated at the connections to the Spent Fuel Pit Cooling System and Temporary Waste Water Processing Island.

227 DEMINERALIZED WATER SYSTEM

227.1 Description

The Demineralized Water System supports plant operations by providing demineralized water for the following major uses:

- Make-up water to Spent Fuel Pit
- Water for decontamination activities

The system consists of a water storage tank, one make-up pump, and the necessary associated valves, piping, fittings, hoses and instrumentation (Drawing Number 9699-FM-101).

The system is required to meet make-up requirements for the Spent Fuel Pit.

The Demineralized Water System may also be used to support dismantlement activities. Demineralized Water System make-up requirements may increase during dismantlement activities to support the filling of systems for shielding as well as increases in decontamination activities.

227.2 Decontamination and Dismantlement Considerations

General decontamination and dismantlement considerations are presented in Section 200.3.

The following is a specific system consideration:

- The Demineralized Water System should be isolated at the connections to the plant systems as they are isolated and dismantled.

228 COMPRESSED AIR SYSTEM

228.1 Description

The Compressed Air System provides air for plant use. The system consists of portable electric and/or diesel-driven air compressors, receiver tanks, and the necessary associated valves, piping, fittings and instrumentation.

The Compressed Air System is required to support dismantlement activities. The system is also needed to provide compressed air to the fuel handling equipment until fuel is permanently removed from the Spent Fuel Pit, and the Temporary Waste Water Processing Island System.

The Compressed Air System and components are located in various areas of the plant. The system should remain in service to support decommissioning activities. Portions of the Compressed Air System will be isolated, dismantled and removed as the systems and areas that they support are dismantled and removed from service.

229 ELECTRICAL SYSTEM

229.1 Description

The on-site electrical system is powered by the 13.8 kV Massachusetts Electric Line. The system consists of transformers, switchboards, motor control centers, distribution panels and the necessary associated instrumentation and controls (Drawing Numbers 9699-FE-1JA, -1P, -1Q, -1R and -1S).

The 13.8 kV Massachusetts Electric Line also powers the Furlon House, Training Center and Trash Compactor. The electrical system at YNPS provides power to equipment which must remain energized during the final phase of decommissioning and while fuel is located in the Spent Fuel Pit. All on-site electrical equipment is powered from two 480 V AC switchboards via two 13.8 kV/480 V, 100 kVA transformers. The 1600 amp Secondary Side Switchboard powers equipment located on the Secondary side of the plant. The Gate House and ISFSI are supplied from the Secondary Side Switchboard. The 1200 amp Primary Side Switchboard powers the equipment located on the Primary side of the plant. The Spent Fuel Pit MCC and the Fuel Transfer Enclosure Switchboard are powered from the Primary Side Switchboard through a manual transfer switch located near the Primary Side Switchboard, which provides for manual transfer of Spent Fuel Pit MCC and Fuel Transfer Enclosure loads to manually started and loaded 600 kW standby diesel generator. Back-up power for portions of the plant electrical system is provided by a manually started loaded 175 kW Security Diesel Generator.

229.2 Decontamination and Dismantlement Considerations

General decontamination and dismantlement considerations are presented in Section 200.3.

230 HEATING SYSTEM

230.1 Description

The Heating System consists of permanent and temporary electric heater units.

The Heating System is required to support plant operations during the dismantlement period. The system is also needed to heat areas of the plant with systems and structures that support fuel storage until fuel is permanently removed from the Spent Fuel Pit.

The Heating System and components are located in various plant buildings. The system should remain operational to support environmental heating requirements during contaminated system removal activities. Temporary heating may be required during area and building dismantlement activities.

231 VENTILATION SYSTEM

231.1 Description

The Ventilation System includes equipment associated with the collection, monitoring, filtration and discharge of potentially radioactive gaseous effluents from specific plant areas.

The Ventilation System (Drawing Number 9699-FM-26D) provides for the controlled airborne ventilation and discharge function. It is used to ventilate and discharge exhaust air via fixed ductwork from the Vapor Container, Spent Fuel Pit (SFP) Building, Fuel Transfer Enclosure and Fan Room. The Ventilation System also ventilates and discharges exhaust air via temporary ducting from the Radioactive Waste Evaporator System, and other areas of the plant as needed to support specific decontamination activities. Essentially, all potentially radioactive airborne effluents are collected by the Ventilation System, filtered through prefilters and High Efficiency Particulate Absolute (HEPA) filters, and discharged through the Primary Vent Stack (PVS). Instrumentation channels monitor the effluent released through the PVS for noble gas, and sample for tritium and particulates.

The requirements to maintain a functional "Ventilation System" to ventilate areas of the plant that support fuel storage or require ventilation during decontamination will exist until all Spent Fuel is removed from the SFP Building, the SFP water has been processed and the associated support buildings and equipment have been decontaminated and ready for dismantlement. In addition, potential airborne releases from these areas shall be processed (filtered) and monitored prior to release as further specified in Yankee Decommissioning Quality Assurance Program (YDQAP) and the Off-Site Dose Calculation Manual (ODCM).

The sample equipment (Drawing Number 9699-FK-13A) includes four isokinetic sample nozzles. There are two sample pathways provided. Each sample pathway draws a sample from one of two isokinetic nozzles to the sample and monitoring equipment located in the Stack Monitoring House on top of the PAB. One nozzle is sized for a nominal stack flow for one fan operation, and the second nozzle sized for a nominal stack flow for two fan operation.

The Ventilation System components and equipment are located in or on the Primary Auxiliary Building, Vapor Container and Yard Area.

231.2 Decontamination and Dismantlement Considerations

General decontamination and dismantlement considerations are presented in Section 200.3. The following are specific system considerations:

- The fans and motors should be separated from the baseplates prior to removal.

- The filter units should be dismantled into manageable sections.

232 FIRE PROTECTION AND DETECTION SYSTEM

232.1 Description

The Fire Protection and Detection System provides the equipment needed to detect and respond to fires that could occur in the plant. The system consists of electric and diesel driven fire pumps, pressure maintenance system, hydrants, hoses, detectors, and the necessary associated valves, piping, fittings and instrumentation (Drawing Numbers 9699-FM-90A and -90C).

Portions of the Fire Protection and Detection System are required to support plant operations during the dismantlement period. The system is also needed to protect areas of the plant with systems and structures that support fuel storage until fuel is permanently removed from the Spent Fuel Pit. The Fire Protection Technical Requirements Manual (FPTRM) presents system availability requirements (Reference 513-2).

The Fire Protection and Detection System and components are located in plant buildings and structures as described in the Fire Protection Technical Requirements Manual.

The system should remain in service until all contaminated process systems have been dismantled and fuel is permanently removed from the Spent Fuel Pit. However, portions of the Fire Protection and Detection System may be disconnected, isolated and removed when they are no longer required to support fire protection requirements.

232.2 Decontamination and Dismantlement Considerations

General decontamiantion and dismantlement considerations are presented in Section 200.3. The following is a specific system consideration:

- Modifications to the Fire Protection and Detection System require review and modifications, as necessary, to the YNPS Fire Protection Plan (Section 513).

233 PRIMARY PUMP SEAL WATER SYSTEM

233.1 Status

The Primary Pump Seal Water System has been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

234 SAFE SHUTDOWN SYSTEM

234.1 Status

The Safe Shutdown System has been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

235 WATER CLEANUP SYSTEM

235.1 Status

The Water Cleanup System has been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

236 VAPOR CONTAINER

236.1 Description

The Vapor Container is a spherical steel structure that surrounds the Reactor Support Structure. It is located about 23 feet above grade and is supported by 16 steel columns. Each column is braced by steel rods that provide cross bracing for lateral loads. The steel columns are supported by reinforced concrete pedestals.

The Vapor Container provides lateral support to the Vapor Container service elevator tower, and the Primary Vent Stack (PVS). Attachments are limited to minor platform framing, exterior stairs, and lightly loaded supports for pipes and cable trays.

The following comprise major Vapor Container penetrations:

- <u>Reactor Support Structure Support Columns</u> - The Reactor Support Structure support columns carry the Reactor Support Structure load. Each column is isolated from the Vapor Container by bellows located at the Vapor Container shell.

- <u>Fuel Transfer Chute</u> - The Fuel Transfer Chute is a concrete shielded pipe connecting the Shield Tank Cavity and the Spent Fuel Pit. The chute was used to transfer fuel assemblies, control rods, and miscellaneous irradiated components. The chute has been decommissioned and a temporary cover placed on the Vapor Container shell.

- <u>Equipment Hatch</u> - The Equipment Hatch provides an opening to the Vapor Container lower hemisphere. The hatch allows transport of equipment into and out of the Vapor Container. A temporary cover may be used to isolate the Vapor Container.

- <u>Personnel Hatch</u> - The Personnel Hatch provided a dual airlock for personnel access into and out of the Vapor Container when containment integrity was required. A temporary door may be used in place of the Personnel Hatch to isolate the Vapor Container.

- <u>Piping Penetrations</u> - Both piping penetrations have been cut out to allow for removal of equipment from the Vapor Container.

- <u>Electrical Penetrations</u> - Electrical penetrations are made from steel pipe penetrations welded into the Vapor Container with bolted and gasketed flanges.

The Vapor Container shell prevents the spread of contamination. Access to the Vapor Container will be controlled to limit access. The Vapor Container shell will be used as a barrier to control the release of radioactive materials that may become airborne during dismantlement activities.

The capability to establish Vapor Container isolation will be maintained during decommissioning activities inside the containment. The Vapor Container is not required to store spent fuel and could be decontaminated and decommissioned; however, the Vapor Container will not be dismantled until the fuel chute has been physically disconnected from the Spent Fuel Pit.

236.2 Decontamination and Dismantlement Considerations

General decontamination and dismantlement considerations are presented in Section 200.3. The following are specific area considerations:

- Piping penetrations should be cut off as close as practicable to the Vapor Container shell when the process system which passes through it is dismantled. The opening in the Vapor Container shell should be closed once the piping is removed.

- Electrical penetrations should be cut off as close as practicable to the Vapor Container shell after all cables in the penetration have been disconnected and removed. The opening in the Vapor Container shell should be closed once the penetration is removed.

- Platforms, ladders, and stairs along with the supporting steel members should be removed in conjunction with area decontamination and dismantlement activities.

- Dismantlement of the Vapor Container shell should not commence until the fuel chute has been physically disconnected from the Spent Fuel Pit.

237 REACTOR SUPPORT STRUCTURE

237.1 Description

The Reactor Support Structure is a reinforced concrete structure which supports the polar crane.

The Reactor Support Structure consists of two concentric reinforced concrete cylinders. The cylinders are connected together with reinforced concrete radial walls which formed compartments for the main coolant loops, Pressurizer and Equipment Hatch.

The compartments are covered by a reinforced concrete charging floor. The charging floor is composed of removable concrete slabs which allow crane access to the compartments.

The Reactor Support Structure is supported on eight reinforced concrete steel encased columns which penetrate the Vapor Container shell. The Vapor Container penetrations are sealed by stainless steel expansion joints. An annular space is provided to permit the Vapor Container and the internal concrete structure to move independently.

237.2 Decontamination and Dismantlement Considerations

General decontamination and dismantlement considerations are presented in Section 200.3. The following are specific area considerations:

- The shield tank cavity liner should be removed to permit decontamination of the underlying concrete surfaces.

- The steel casings of the support columns from the shell to the expansion joint should be removed to permit access to the concrete column.

- The concrete columns should be decontaminated by removing the contaminated concrete.

- All contaminated equipment should be removed prior to decontamination or removal of concrete on the walls, floors and ceilings.

- Concrete and reinforcing bar on the inner section of the inner support wall, which was behind the Neutron Shield Tank, is slightly activated and may require partial removal.

- The concrete and reinforcing around the main coolant loop penetrations may also be slightly activated. The removal zone will be determined using cored samples of the concrete and reinforcing.

238 VAPOR CONTAINER POLAR CRANE

238.1 Description

The Vapor Container Polar Crane was used to support refueling and maintenance related activities inside the Vapor Container. The crane was originally designed for the installation of the Reactor Vessel and Steam Generators. However, crane capacity was reduced during plant operations by converting one hook to a smaller capacity to increase hook travel speed. The smaller hook was replaced with a larger hook as part of the Component Removal Project, returning the polar crane to its original capacity. After removal of the Reactor Pressure Vessel, the larger hook was again replaced with a smaller hook.

The crane consists of a bridge which rides on a 75 foot diameter crane rail with a common trolley rigged with two hooks. The rated capacity of the bridge and common trolley is 150 ton. The installed hooks have rated capacities of 75 ton (Hook No. 1) and 15 ton (Hook No. 2).

The Vapor Container Polar Crane will be used to support decontamination and dismantlement activities in the Vapor Container.

238.2 Decontamination and Dismantlement Considerations

General decontamination and dismantlement considerations are presented in Section 200.3. The following are specific area considerations:

- The Vapor Container Polar Crane should be decontaminated at the same time as the Vapor Container shell or removed from the Vapor Container and decontaminated at a designated decontamination area/facility.

- The hoist trolley, motors, and control cab should be removed from the girders.

239 RADIATION SHIELDING

239.1 Description

Radiation shielding is installed for both personnel and equipment protection. The radiation shielding is comprised of several categories according to function:

- Primary Shield - The Primary Shield was a cylindrical, reinforced concrete structure immediately adjacent to and surrounding the exterior of the Neutron Shield Tank. Dismantlement of the Primary Shield has been initiated by the mechanical removal of portions of the reinforced concrete.

  The lower portion of the primary shield is 4.5 to 5 feet thick and extends from the Reactor Support Structure to the Shield Tank Cavity floor. The upper portion of the shield forms the walls of the Shield Tank Cavity. The Primary Shield is 6 feet thick in the area adjacent to the Equipment Hatch.

- Secondary Shield - The Secondary Shield surrounded the entire primary plant inside the Vapor Container. The shield reduced radiation dose outside of the Vapor Container to below acceptable levels. The bottom portion of the shield forms a section of the main structural support for the primary plant. The portion of the shield below the charging floor is 5 feet thick. The structure also shielded portions of the Steam Generator above the charging floor. The upper portion of the secondary shield is 2 feet thick and extends above the charging floor to support the polar crane.

  Additional shield walls located between the primary and secondary shields segregated the main coolant loop, the Steam Generators, the Pressurizer, and the Equipment Hatch compartments. The partitions separating the main coolant loops and Steam Generators are 2 feet thick. The Pressurizer partitions are 1.5 feet thick. The walls surrounding the Equipment Hatch taper from 5 feet thick at the base to 2 feet thick at the top.

- Auxiliary Shielding - Auxiliary shielding protects personnel during decontamination and dismantlement activities in several plant areas. This shielding will be maintained as necessary during the dismantlement periods to prevent inadvertent exposure to radiation.

239.2 Decontamination and Dismantlement Considerations

General decontamination and dismantlement considerations are presented in Section 200.3. The following are specific area considerations:

- Auxiliary shielding will be decontaminated and dismantled as part of the area and building decontamination and dismantlement activity.

- Supplemental shielding may be decontaminated and dismantled at any time.

240 NEUTRON SHIELD TANK

240.1 Status

The Neutron Shield Tank has been permanently removed. This section of the FSAR is no longer applicable to the YNPS and has been deleted.

241 PIPE CHASES

241.1 Description

There are two pipe chases between the Primary Auxiliary Building and the Vapor Container:

- Lower Pipe Chase - The Lower Pipe Chase is a corridor that runs between the second story of the Primary Auxiliary Building and the Vapor Container lower hemisphere. The chase is constructed of reinforced concrete.

- Upper Pipe Chase - The Upper Pipe Chase is a corridor that runs from the Primary Auxiliary Building roof to the Vapor Container lower hemisphere. The chase is constructed of concrete masonry units and is supported by the lower pipe chase.

The piping in both pipe chases has been removed and the Vapor Container shell cut to allow for easier removal of equipment and components from the Vapor Container, and also serve as an alternate personnel access to the Vapor Container.

242 FUEL TRANSFER CHUTE

242.1 Description

The Fuel Transfer Chute was used to transfer new and spent fuel, as well as irradiated components, between the Spent Fuel Pit and the Vapor Container. The chute was a series of stainless steel pipe sections connected by bolted flanges enclosed in a reinforced concrete tunnel. The chute is structurally isolated from the Vapor Container by a metal bellows expansion joint. The Fuel Transfer Chute was accessed through a below grade manhole tank.

The integrity of the Fuel Transfer Chute is necessary to maintain the Spent Fuel Pit integrity. The Fuel Transfer Chute has been isolated by: (a) resupporting the Fuel Transfer Chute/Spent Fuel Pit penetration assembly to the Spent Fuel Pit using the latch mechanism, (b) filling the annular space between the Fuel Transfer Chute pipe and the Spent Fuel Pit penetration pipe with grout, (c) removing one section of Fuel Transfer Chute pipe uphill of the Lower Lock Valve (LLV), (d) installing a blind flange cap on the LLV, (e) erecting permanent form work and placing a concrete barrier in the LLV pit and (f) installing metal plates above and below the LLV pit to preclude personnel access to this area. This has been completed with the implementation of EDC 95-303, "Fuel Transfer Chute Isolation."

The stainless steel pipe sections above the LLV have been removed. The LLV is significantly contaminated. Removal of the Fuel Transfer Chute and LLV from the Spent Fuel Pit wall can only be performed after the fuel has been removed from the pit.

242.2 Decontamination and Dismantlement Considerations

General decontamination and dismantlement considerations are presented in Section 200.3. The following are specific area considerations:

- The Fuel Transfer Chute should be dismantled as follows:
  - Coat the internal surfaces of the Fuel Transfer Chute Structure and entrance manhole with a fixative to bind the loose contamination.
  - Remove the removable shield blocks from the access openings on top of the Fuel Transfer Chute Structure. Decontaminate the blocks as required.
  - Remove the lead shielding from the Fuel Transfer Chute Structure at the Vapor Container shell penetration.
  - Encase the lower lock valve in concrete to provide shielding and protection for the valve and Fuel Transfer Chute to the Spent Fuel Pit (Figure 242-1).
  - Segment the Fuel Transfer Chute and Structure into manageable sections using a diamond wire saw or a similar method. Start segmenting at the Vapor Container shell and stop as close as practicable to the lower lock valve concrete enclosure.

- Remove the section of concrete which encases the lower lock valve and assure that any residual water remaining in the lower lock valve is removed prior to disposal.

- Remove the Fuel Transfer Chute/Spent Fuel Pit penetration as a unit and place it into a shipping cask for disposal.

- Excavate the entrance manhole and move it to a cutting station for segmentation.

- Close the opening in the Vapor Container shell at the location of the Fuel Transfer Chute penetration.

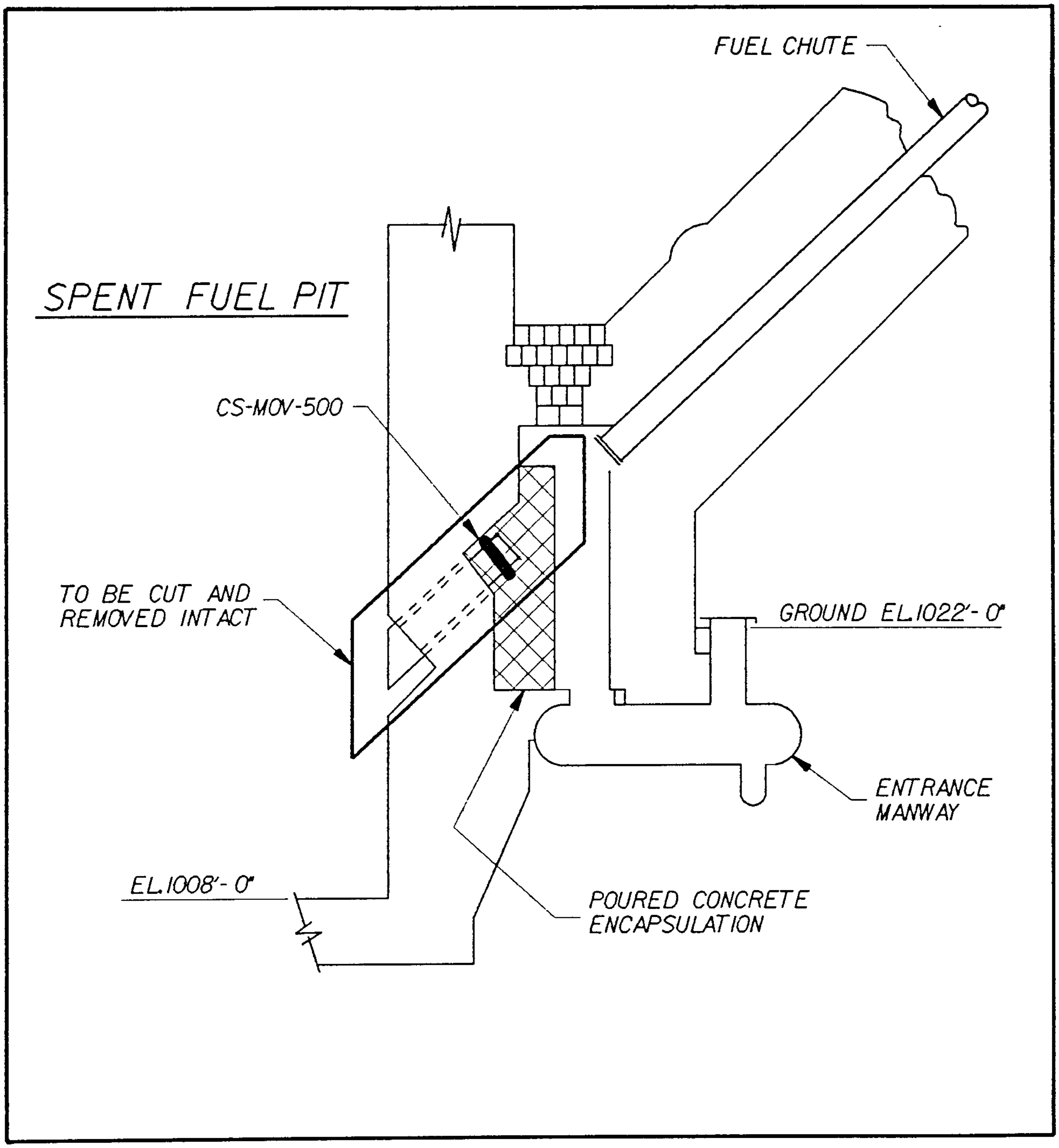


FIGURE 242-1

Spent Fuel Pit and Fuel Chute Removal Detail

243 YARD AREA CRANE AND SUPPORT STRUCTURE

243.1 Description

The Yard Area Crane Support Structure is a braced steel frame structure that supports a crane that services the Ion Exchange Pit, Spent Fuel Pit and Decontamination Room. The crane support structure is approximately 34 feet by 151 feet by 73 feet high with a design capacity of 80 tons.

The Yard Area Crane was upgraded to a single-failure-proof crane meeting the requirements of NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants" with the implementation of EDC 96-305, "Yard Area Crane Upgrades." This upgrade also increased the capacity of the Yard Area Crane from 75 tons to 80 tons which also necessitated reinforcement of the Yard Area Crane Support Structure.

The Yard Area Crane upgrade design employs the manufacturer's [Ederer, Inc., Seattle, WA] Generic Licensing Topical Report EDR-1(P), Revision 3, "Ederer Nuclear Safety-Related e<u>X</u>tra <u>S</u>afety <u>A</u>nd <u>M</u>onitoring (X-SAM) Cranes." EDR-1(P) documents the NRC review and generic licensing acceptance of Nuclear Safety Related X-SAM Cranes for use in single-failure-proof applications in existing facilities. Plant-specific details, as delineated in Appendices B and C of EDR-1(P), are presented in Attachments 243-1 and 243-2, respectively.

The Yard Area Crane and Support Structure will be used to support fuel management activities until fuel is permanently removed from the Spent Fuel Pit. The crane will also be used to support activities associated with the Spent Fuel Pit, Ion Exchange Pit and other heavy lifts.

REFERENCES

243-1 Letter, M. B. Fairtile (NRC) to F. N. Williams (YAEC), "Issuance of Amendment No. 149 to Facility Operating License (Possession Only) No. DPR-3, Yankee Nuclear Power Station," June 17, 1998.

243-2 EDR-1(P), Generic Licensing Topical Report EDR-1(P), Revision 3, "Ederer Nuclear Safety-Related eXtra Safety And Monitoring (X-SAM) Cranes."

ATTACHMENT 243-1

APPENDIX B SUPPLEMENT TO GENERIC LICENSING TOPICAL REPORT EDR-1

SUMMARY OF PLANT SPECIFIC CRANE DATA SUPPLIED BY EDERER INCORPORATED FOR YANKEE ATOMIC ELECTRIC COMPANY YANKEE NUCLEAR POWER STATION YARD AREA CRANE

P.O. NO. QA42114

EDERER S..O. NO. F2592

REVISION 0

7/8/98

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PROJECT ENGINEER
EDERER INCORPORATED

CHECKED: ____________
EDERER INCORPORATED

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EDERER INCORPORATED

APPROVED: ____________
CHIEF ENGINEER
EDERER INCORPORATED

ATTACHMENT 243-1
(Continued)

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EDR-1 APPENDIX B SUPPLEMENT
SUMMARY OF PLANT SPECIFIC CRANE DATA SUPPLIED BY EDERER FOR
YANKEE NUCLEAR POWER STATION YARD AREA CRANE
EDERER S.O. NO. F2592

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EDR-1 APPENDIX B SUPPLEMENT
SUMMARY OF PLANT SPECIFIC CRANE DATA SUPPLIED BY EDERER FOR
YANKEE NUCLEAR POWER STATION YARD AREA CRANE

REGULATORY POSITION	TOPICAL REPORT SECTION		INFORMATION TO BE PROVIDED		SPECIFIC CRANE DATA
C.1.a	III.C (C.1.a)	1.	THE ACTUAL CRANE DUTY CLASSIFICATION OF THE CRANE SPECIFIED BY THE APPLICANT	1.	THE CRANE HAS A CLASS "A" CRANE DUTY CLASSIFICATION IN ACCORDANCE WITH CMAA SPECIFICATION #70.
C.1.b	III.C (C.1.b)	1.	THE MINIMUM OPERATING TEMPERATURE OF THE CRANE SPECIFIED BY THE APPLICANT.	1.	THE TROLLEY WAS DESIGNED AND FABRICATED FOR A MINIMUM-OPERATING TEMPERATURE OF 0°F.
C.2.b	III.C (C.2.b) III.E.4	1.	THE MAXIMUM EXTENT OF LOAD MOTION AND THE PEAK KINETIC ENERGY OF THE LOAD FOLLOWING A DRIVE TRAIN FAILURE	1.	THE MAIN HOIST WAS DESIGNED SUCH THAT THE MAXIMUM LOAD FOLLOWING A DRIVE TRAIN FAILURE IS LESS THAN 1 FOOT AND THE MAXIMUM KINETIC ENERGY OF THE LOAD IS LESS THAN THAT RESULTING FROM 1 INCH OF FREE FALL OF THE MAXIMUM CRITICAL LOAD.
		2.	PROVISIONS FOR THE ACTUATING THE EMERGENCY DRUM BRAKE PRIOR TO TRAVERSING WITH THE LOAD, WHEN REQUIRED TO ACCOMMODATE THE LOAD MOTION FOLLOWING A DRIVE TRAIN FAILURE.	2.	PROVISIONS FOR AUTOMATICALLY ACTUATING THE EMERGENCY DRUM BRAKE PRIOR TO TRAVERSING WITH THE LOAD ARE NOT REQUIRED SINCE PLANT PROCEDURES WILL LIMIT THE HEIGHT OF CRITICAL LIFTS TO BETWEEN 13" AND 18" ABOVE THE OPERATING FLOOR OF THE SPENT FUEL POOL BUILDING.

ATTACHMENT 243-1
(Continued)

EDR-1 APPENDIX B SUPPLEMENT
SUMMARY OF PLANT SPECIFIC CRANE DATA SUPPLIED BY EDERER FOR YANKEE NUCLEAR POWER STATION YARD AREA CRANE

REGULATORY POSITION	TOPICAL REPORT SECTION		INFORMATION TO BE PROVIDED		SPECIFIC CRANE DATA
C.3.e	III.C(C.3.e)	1.	THE MAXIMUM CABLE LOADING FOLLOWING A WIRE ROPE FAILURE IN TERMS OF THE ACCEPTANCE CRITERIA ESTABLISHED IN SECTION III.C (C.3.e)	1.	THE REQUIRED STRENGTH BREAKING FOLLOWING A WIRE ROPE FAILURE IN THE MAIN HOIST IS 160K. THE BREAKING STRENGTH PROVIDED =175.8K.
C.3.f	--	1.	MAXIMUM FLEET ANGLE	1.	3.5 DEGREES.
		2.	NUMBER OF REVERSE BENDS	2.	NONE, OTHER THAN THE ONE BETWEEN THE WIRE ROPE DRUM AND THE FIRST SHEAVE IN THE LOAD BLOCK.
		3.	SHEAVE DIAMETER	3.	20 IN MINIMUM PER CMAA SPECIFICATION #70.
C.3.h	III.C (C.3.h) III.E.11	1.	THE MAXIMUM EXTENT OF MOTION AND PEAK KINETIC ENERGY OF THE LOAD FOLLOWING A SINGLE WIRE ROPE FAILURE.	1.	THE MAIN HOIST WAS DESIGNED SUCH THAT THE MAXIMUM LOAD MOTION FOLLOWING A SINGLE WIRE ROPE FAILURE IS LESS THAN 1 FOOT AND THE MAXIMUM KINETIC ENERGY OF THE LOAD IS LESS THAN THAT RESULTING FROM ONE INCH OF FREE FALL OF THE MAXIMUM CRITICAL LOAD = 13,330 LB.FT. CALCULATED MAX LOAD MOTION=0.449 FT. CALCULATED K.E. = 6598 LB.FT.

ATTACHMENT 243-1
(Continued)

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EDR-1 APPENDIX B SUPPLEMENT
SUMMARY OF PLANT SPECIFIC CRANE DATA SUPPLIED BY EDERER FOR YANKEE NUCLEAR POWER STATION YARD AREA CRANE

REGULATORY POSITION	TOPICAL REPORT SECTION	INFORMATION TO BE PROVIDED	SPECIFIC CRANE DATA
C.3.i	III.C (C.3.i)	1. THE TYPE OF LOAD CONTROL SYSTEM SPECIFIED BY THE APPLICANT.	1. EDERER AC FLUX VECTOR
		2. WHETHER INTERLOCKS ARE RECOMMENDED BY REGULATORY GUIDE 1.13 TO PREVENT TROLLEY AND BRIDGE MOVEMENTS WHILE FUEL ELEMENTS ARE BEING LIFTED AND WHETHER THEY ARE PROVIDED FOR THIS APPLICATION.	2. THE CRANE WILL NOT BE USED TO LIFT FUEL ELEMENTS FROM THE REACTOR CORE OR SPENT FUEL RACKS. HOWEVER A BRIDGE ZONE LIMIT SWITCH WILL PREVENT LOADS FROM BEING MOVED OVER THE FUEL RACKS.

ATTACHMENT 243-1
(Continued)

EDR-1 APPENDIX B SUPPLEMENT
SUMMARY OF PLANT SPECIFIC CRANE DATA SUPPLIED BY EDERER FOR
YANKEE NUCLEAR POWER STATION YARD AREA CRANE

REGULATORY POSITION	TOPICAL REPORT SECTION	INFORMATION TO BE PROVIDED	SPECIFIC CRANE DATA
C.3.j	III.C (C.3.j)	1. THE MAXIMUM CABLE AND MACHINERY LOADING THAT WOULD RESULT IN THE EVENT OF A HIGH SPEED TWO BLOCKING, ASSUMING A CONTROL SYSTEM MALFUNCTION THAT WOULD ALLOW THE FULL BREAKDOWN TORQUE OF THE MOTOR TO BE APPLIED TO THE DRIVE MOTOR SHAFT.	1. THE ENERGY ABSORBING TORQUE LIMITER (EATL) WAS DESIGNED SUCH THAT THE MAXIMUM MACHINERY LOAD, WHICH WOULD RESULT IN THE EVENT A TWO BLOCKING OCCURS WHILE LIFTING THE RATED LOAD AT THE RATED SPEED AND THAT ALLOWS THE FULL BREAKDOWN TORQUE OF THE MOTOR TO BE APPLIED TO THE DRIVE SHAFT, WILL NOT EXCEED 3 TIMES THE DESIGN RATED LOADING. IN ADDITION, THE EATL DESIGN DOES NOT ALLOW THE MAXIMUM CABLE LOADING TO EXCEED THE ACCEPTANCE CRITERIA ESTABLISHED IN SECTION III.C (C.3.e) DURING THE ABOVE DESCRIBED TWO-BLOCKINGS.
		2. MEANS OF PREVENTING TWO BLOCKING OF AUXILIARY HOIST, IF PROVIDED.	2. THE AUXILIARY HAS A ROTARY CONTROL TYPE LIMIT SWITCH AS THE FIRST PRIMARY LIMIT AND A BLOCK ACTUATION PLUGGING CIRCUIT TYPE LIMIT SWITCH THAT COMMANDS THE HOIST TO LOWER THE BLOCK AS A SECONDARY LIMIT.

ATTACHMENT 243-1
(Continued)

EDR-1 APPENDIX B SUPPLEMENT
SUMMARY OF PLANT SPECIFIC CRANE DATA SUPPLIED BY EDERER FOR
YANKEE NUCLEAR POWER STATION YARD AREA CRANE

REGULATORY POSITION	TOPICAL REPORT SECTION	INFORMATION TO BE PROVIDED	SPECIFIC CRANE DATA
C.3.k	III.C(C.3.k)	1. TYPE OF DRUM SAFETY SUPPORT PROVIDED.	1. THE ALTERNATE DESIGN DRUM SAFETY RESTRAINT SHOWN IN FIGURE III.D.4 OF EDR-1 IS ARRANGED TO COUNTER GEAR AND BRAKE FORCES AS WELL AS DOWNWARD LOADS. THESE SUPPORTS ACT ON THE ENDS OF THE DRUM.
C.3.o	--	1. TYPE OF HOIST DRIVE TO PROVIDE INCREMENTAL	1. 50:1 SLOW SPEED IS PROVIDED AS A PART OF AC FLUX VECTOR CONTROL.
C.3.p	--	1. MAXIMUM TROLLEY SPEED	1. 50 F.P.M.
		2. MAXIMUM BRIDGE SPEED	2. 50 F.P.M.
		3. TYPE OF OVERSPEED PROTECTION FOR THE TROLLEY AND BRIDGE DRIVES.	3. OVERSPEED SWITCH DRIVEN FROM MOTOR.

ATTACHMENT 243-1
(Continued)

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EDR-1 APPENDIX B SUPPLEMENT
SUMMARY OF PLANT SPECIFIC CRANE DATA SUPPLIED BY EDERER FOR YANKEE NUCLEAR POWER STATION YARD AREA CRANE

REGULATORY POSITION	TOPICAL REPORT SECTION	INFORMATION TO BE PROVIDED	SPECIFIC CRANE DATA
C.3.q	--	1. CONTROL STATION LOCATION	1. THE COMPLETE OPERATING CONTROL SYSTEM, INCLUDING THE EMERGENCY STOP BUTTONS, ARE LOCATED IN THE BRIDGE MOUNTED CRANE CAB.
--	III.D.1	1. THE TYPE OF EMERGENCY DRUM BRAKE USED, INCLUDING TYPE OF RELEASE MECHANISM.	1. A SINGLE PNEUMATICALLY RELEASED BAND BRAKE WILL BE USED.
		2. THE RELATIVE LOCATION OF THE EMERGENCY DRUM BRAKE.	2. THE EMERGENCY DRUM BRAKE ENGAGES THE MAIN HOIST WIRE ROPE DRUM.
		3. EMERGENCY DRUM BRAKE CAPACITY.	3. THE MAIN HOIST EMERGENCY DRUM BRAKE HAS A MINIMUM CAPACITY OF 130% OF THAT REQUIRED TO HOLD THE DESIGN RATED LOAD.
--	III.D.2	1. NUMBER OF FRICTION SURFACES	1. THE EATL HAS 17 FRICTION SURFACES.
		2. EATL TORQUE SETTING	2. THE SPECIFIED EATL TORQUE SETTING IS APPROXIMATELY 130% OF THE MAIN HOIST DESIGN RATED LOAD.

ATTACHMENT 243-1
(Continued)

EDR-1 APPENDIX B SUPPLEMENT
SUMMARY OF PLANT SPECIFIC CRANE DATA SUPPLIED BY EDERER FOR YANKEE NUCLEAR POWER STATION YARD AREA CRANE

REGULATORY POSITION	TOPICAL REPORT SECTION	INFORMATION TO BE PROVIDED	SPECIFIC CRANE DATA
–	III.D.3	1. TYPE OF FAILURE DETECTION SYSTEM.	1. A TOTALLY MECHANICAL DRIVE TRAIN CONTINUITY DETECTOR AND EMERGENCY DRUM BRAKE ACTUATOR HAVE BEEN PROVIDED IN ACCORDANCE WITH APPENDIX G OF REVISION 3 OF EDR-1 FOR THE MAIN HOIST.
–	III.D.5	1. TYPE OF HYDRAULIC LOAD EQUALIZATION SYSTEM.	1. THE MAIN HOIST HYDRAULIC LOAD EQUALIZATION SYSTEM INCLUDES BOTH FEATURES DESCRIBED IN SECTION III.D.5.
–	III.D.6	1. TYPE OF HOOK.	1. THE MAIN HOOK IS A SINGLE LOAD PATH SISTER HOOK.
		2. HOOK DESIGN LOAD	2. THE MAIN HOOK DESIGN CRITICAL LIFT LOAD IS 80 TONS WITH A 10:1 FACTOR OF SAFETY ON ULTIMATE.
		3. HOOK TEST LOAD	3. THE TEST LOAD FOR THE MAIN HOOK IS 200% OF THE CRITICAL LOAD (I.E. 200% OF 80 TONS=160 TONS)

ATTACHMENT 243-1
(Continued)

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EDR-1 APPENDIX B SUPPLEMENT
SUMMARY OF PLANT SPECIFIC CRANE DATA SUPPLIED BY EDERER FOR YANKEE NUCLEAR POWER STATION YARD AREA CRANE

REGULATORY POSITION	REPORT SECTION	INFORMATION TO BE PROVIDED	SPECIFIC CRANE DATA
--	III.F.1	1. DESIGN RATED LOAD.	1. MAIN HOIST - 80 TONS
		2. MAXIMUM CRITICAL LOAD RATING.	2. MAIN HOIST - 80 TONS
		3. TROLLEY WEIGHT (NET).	3. 41,000 LBS. (INCLUDING HOOKS)
		4. TROLLEY WEIGHT (WITH LOAD)	4. 201,000 LBS.
		5. HOOK LIFT.	5. MAIN HOOK - 68 FEET, 6 INCHES
		6. NUMBER OF WIRE ROPE DRUMS	6. THE MAIN HOIST HAS ONE WIRE ROPE DRUM.
		7. NUMBER OF PARTS OF WIRE.	7. MAIN HOIST - 4 PARTS PER WIRE ROPE.
		8. DRUM SIZE (PITCH DIAMETER).	8. MAIN HOIST - 29 ¾ INCHES
		9. WIRE ROPE DIAMETER	9. MAIN HOIST - 1 1/4 INCH
		10. WIRE ROPE TYPE.	10. 6x37 CLASS EEIPS/IWRC - MAIN HOIST
		11. WIRE ROPE MATERIAL.	11. CARBON STEEL - MAIN HOIST
		12. WIRE ROPE BREAKING STRENGTH.	12. MAIN HOIST - 175,800 LBS.
		13. WIRE ROPE YIELD STRENGTH	13. MAIN HOIST - 140,600 LBS.
		14. WIRE ROPE RESERVE STRENGTH. (% OF BREAKING STRENGTH)	14. MAIN HOIST - 56.4%
		15. NUMBER OF WIRE ROPES.	15. THE MAIN HOIST HAS TWO ROPES.

ATTACHMENT 243-2

APPENDIX C SUPPLEMENT TO
GENERIC LICENSING TOPICAL REPORT
EDR-1

SUMMARY OF PLANT SPECIFIC CRANE DATA
SUPPLIED BY APPLICANT
FOR
YANKEE ATOMIC ELECTRIC COMPANY
YANKEE NUCLEAR POWER STATION
YARD AREA CRANE

P.O. NO. QA42114

EDERER S..O. NO. F2592

REVISION 0

7/8/98

PREPARED: ______
PROJECT ENGINEER
EDERER INCORPORATED

CHECKED: ______
EDERER INCORPORATED

REVIEWED: ______
QUALITY ASSURANCE MGR.
EDERER INCORPORATED

APPROVED: ______
CHIEF ENGINEER
EDERER INCORPORATED

ATTACHMENT 243-2
(Continued)

REVISION O

EDR-1 APPENDIX C SUPPLEMENT
SUMMARY OF REGULATORY POSITIONS TO BE ADDRESSED BY THE APPLICANT
YANKEE NUCLEAR POWER STATION YARD AREA CRANE
EDERER S.O. NO. F2592

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(Continued)

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EDR-1 APPENDIX C SUPPLEMENT
SUMMARY OF REGULATORY POSITIONS TO BE ADDRESSED BY THE APPLICANT
YANKEE NUCLEAR POWER STATION YARD AREA CRANE

REGULATORY POSITION	TOPICAL REPORT SECTION	INFORMATION TO BE PROVIDED	SPECIFIC CRANE DATA
--	III.C(C.1.b.(1))	1. THE EXTENT OF VENTING OF CLOSED BOX SECTIONS.	1. CLOSED BOX SECTIONS ARE NOT VENTED SINCE THE YARD AREA CRANE IS NOT IN PRESSURIZED SPACE.
C.1.b(3) C.1.b(4) C.4.d	III.C(C.1.b(3)) III.C(C.1.b(4)) III.C(C.4.d)	1. THE NONDESTRUCTIVE AND COLD PROOF TESTING TO BE PERFORMED ON EXISTING STRUCTURAL MEMBERS FOR WHICH SATISFACTORY IMPACT TEST DATA IS NOT AVAILABLE.	1. NOT APPLICABLE AS NO EXISTING CRANE COMPONENTS ARE REUSED.
C.1.c	III.C(C.1.c)	1. THE EXTENT THE CRANE'S STRUCTURES WHICH ARE NOT BEING REPLACED ARE CAPABLE OF MEETING THE SEISMIC REQUIREMENTS OF REGULATORY GUIDE 1.29.	1. NOT APPLICABLE AS NO EXISTING CRANE COMPONENTS ARE REUSED.
C.1.d	III.C(C.1.d)	1. THE EXTENT WELD JOINTS IN THE CRANE'S STRUCTURES, WHICH ARE NOT BEING REPLACED, WERE NONDESTRUCTIVELY EXAMINED.	1. NOT APPLICABLE. THE CRANE'S STRUCTURES ARE COMPLETELY REPLACED.

ATTACHMENT 243-2
(Continued)

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C2

EDR-1 APPENDIX C SUPPLEMENT
SUMMARY OF REGULATORY POSITIONS TO BE ADDRESSED BY THE APPLICANT
YANKEE NUCLEAR POWER STATION YARD AREA CRANE

REGULATORY POSITION	TOPICAL REPORT SECTION	INFORMATION TO BE PROVIDED	SPECIFIC CRANE DATA
C.1.d	III.C(C.1.d)	1. THE EXTENT THE BASE MATERIAL, AT JOINTS SUSCEPTIBLE TO LAMELLAR TEARING, WAS NONDESTRUCTIVELY EXAMINED.	1. NOT APPLICABLE.
C.1.e	III.C(C.1e)	1. THE EXTENT THE CRANE'S STRUCTURES, WHICH ARE NOT BEING REPLACED ARE CAPABLE OF WITHSTANDING THE FATIGUE EFFECTS OF CYCLIC LOADING FROM PREVIOUS AND PROJECTED USAGE INCLUDING ANY CONSTRUCTION USAGE.	1. NOT APPLICABLE THE CRANE'S STRUCTURES ARE COMPLETELY REPLACED.
C.1.f	III.C(C.1.f)	1. THE EXTENT THE CRANE'S STRUCTURES WHICH ARE NOT BEING REPLACED, WERE POST-WELD HEAT- TREATED IN ACCORDANCE WITH SUB ARTICLE 3.9 OF AWS D1.1, "STRUCTURAL WELDING CODE".	1. NOT APPLICABLE THE CRANE'S STRUCTURES ARE COMPLETELY REPLACED.

ATTACHMENT 243-2
(Continued)

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EDR-1 APPENDIX C SUPPLEMENT
SUMMARY OF REGULATORY POSITIONS TO BE ADDRESSED BY THE APPLICANT
YANKEE NUCLEAR POWER STATION YARD AREA CRANE

TOPICAL REGULATORY POSITION	REPORT SECTION	INFORMATION TO BE PROVIDED	SPECIFIC CRANE DATA
C.2.b	III.C(C.2.b)	1. PROVISIONS FOR ACCOMMODATING THE LOAD MOTION AND KINETIC ENERGY FOLLOWING A DRIVE TRAIN FAILURE WHEN THE LOAD IS BEING TRAVERSED AND WHEN IT IS BEING RAISED OR LOWERED.	1. THE SPENT FUEL POOL FLOOR IS CAPABLE OF WITHSTANDING THE IMPACT ENERGY RESULTING FROM A 1" FREE FALL DURING LIFTING OR LOWERING OF THE MAXIMUM CRITICAL LOAD (13,333 FT.LBS) WITHOUT A LOSS OF INTEGRITY OF THE LINER. CRANE OPERATING PROCEDURES WILL ENSURE THAT THE LOAD IS LIFTED A MINIMUM OF 13" AND A MAXIMUM OF 18" BEFORE THE LOAD IS TRAVERSED ACROSS THE SPENT FUEL POOL BUILDING FLOOR (EL. 1045') TO ACCOMMODATE 1" OF FREE FALL PLUS THE LOAD MOTION COMMITTED TO FOR FAILURE OF ONE OF THE TWO REDUNDANT REEVING SYSTEMS
C.2.c	III.C(C.2.c)	1. LOCATION OF SAFE LAYDOWN AREAS FOR USE IN THE EVENT REPAIRS TO THE CRANE ARE REQUIRED THAT CANNOT BE MADE WITH THE LOAD SUSPENDED.	1. THERE ARE TWO SAFE LAYDOWN AREAS AVAILABLE FOR USE IN THE EVENT REPAIRS TO THE CRANE ARE REQUIRED THAT CANNOT BE MADE WITH THE LOAD SUSPENDED. THE LOADING/UNLOADING AREA NORTH OF THE SPENT FUEL POOL BUILDING AND SOUTH OF THE DECONTAMINATION BUILDING IS DIMENSIONALLY AND STRUCTURALLY CAPABLE (ON GRADE) AS A LAYDOWN AREA FOR MAXIMUM CRITICAL LOADS. THE SOUTH DECONTAMINATION ROOM IS STRUCTURALL CAPABLE (ON GRADE) OF SUPPORTING THE MAXIMUM CRITICAL LOAD. ACCESS TO THE SOUTH DECONTAMINATION ROOM IS LIMITED TO THE SIZE OF THE ROOF HATCH OPENING. [APPROXIMATELY 12FT X 10.5FT].

ATTACHMENT 243-2
(Continued)

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EDR-1 APPENDIX C SUPPLEMENT
SUMMARY OF REGULATORY POSITIONS TO BE ADDRESSED BY THE APPLICANT
YANKEE NUCLEAR POWER STATION YARD AREA CRANE

TOPICAL REGULATORY POSITION	REPORT SECTION	INFORMATION TO BE PROVIDED	SPECIFIC CRANE DATA
C.2.d	III.C(C.2.d)	1. SIZE OF REPLACEMENT COMPONENTS THAT CAN BE BROUGHT INTO THE BUILDING FOR REPAIR OF THE CRANE WITHOUT HAVING TO BREAK ITS INTEGRITY.	1. NOT APPLICABLE CRANE IS NOT LOCATED INSIDE A BUILDING.
		2. LOCATION OF AREA WHERE REPAIR WORK CAN BE ACCOMPLISHED ON THE CRANE WITHOUT AFFECTING THE SAFE SHUT-DOWN CAPABILITY OF THE REACTOR.	2. NOT APPLICABLE. THE REACTOR HAS BEEN PHYSICALLY REMOVED FROM THE VAPOR CONTAINER (REACTOR CONTAINMENT)
C.2.d	III.C(C.2.d)	3. ANY LIMITATIONS ON REACTOR OPERATIONS THAT WOULD RESULT FROM CRANE REPAIRS.	3. NOT APPLICABLE. THE REACTOR HAS BEEN PHYSICALLY REMOVED FROM THE VAPOR CONTAINER (REACTOR CONTAINMENT)
C.3.b	III.C(C.3.b)	1. THE DESIGN MARGIN AND TYPE OF LIFTING DEVICES THAT ARE ATTACHED TO THE HOOK TO CARRY CRITICAL LOADS.	1. AS AN ALTERNATIVE TO A DUAL LOAD PATH SYSTEM, THE NORMAL STRESS DESIGN FACTORS FOR THE MAIN HOOK HAVE BEEN DOUBLED. EACH LIFTING DEVICE ATTACHED TO THE HOOK TO CARRY CRITICAL LOADS WILLSUPPORT A LOAD SIX TIMES THE STATIC PLUS DYNAMIC LOAD BEING HANDLED WITHOUT PERMANENT DEFORMATION. THE SAFETY FACTOR IS 10:1 WHEN COMPARED TO ULTIMATE. THIS IS ACCORDANCE WITH NUREG 0612, SECTION 5.1.6, PARAGRAPH 1(A) AND ANSI N1 4.6, 1986 SECTION 7.2.1.

ATTACHMENT 243-2
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EDR-1 APPENDIX C SUPPLEMENT
SUMMARY OF REGULATORY POSITIONS TO BE ADDRESSED BY THE APPLICANT
YANKEE NUCLEAR POWER STATION YARD AREA CRANE

REGULATORY POSITION	TOPICAL REPORT SECTION	INFORMATION TO BE PROVIDED	SPECIFIC CRANE DATA
C.3.t	III.C(C.3.t)	1. THE EXTENT CONSTRUCTION REQUIREMENTS FOR THE CRANE'S STRUCTURES, WHICH WILL NOT BE REPLACED, ARE MORE SEVERE THAN THOSE FOR PERMANENT PLANT SERVICE.	1. NOT APPLICABLE
		2. THE MODIFICATIONS AND INSPECTIONS TO BE ACCOMPLISHED ON THE CRANE FOLLOWING CONSTRUCTION USE, WHICH WAS MORE SEVERE THAN THOSE FOR PERMANENT PLANT SERVICE.	2. NOT APPLICABLE
C.3.u	--	1. THE EXTENT OF INSTALLATION AND OPERATING INSTRUCTIONS.	1. THE INSTALLATION AND OPERATING INSTRUCTIONS WILL BE UPDATED BY EDERER TO FULLY COMPLY WITH THE REQUIREMENTS OF SECTION C.3.u OF REGULATORY GUIDE 1.104 AND SECTIONS 7.1 AND 9 OF NUREG-0554. IN ADDITION TO THE EDERER SUPPLIED OPERATING INSTRUCTIONS, OPERATION OF THE CRANE WILL BE IN STRICT CONFORMANCE WITH PLANT APPROVED OPERATIN PROCEDURES GOVERNING THE SAFE HANDLING OF LOADS AND COMPLIANCE WITH LICENSING COMMITMENTS AND TECHNICAL SPECIFICATIONS.

ATTACHMENT 243-2
(Continued)

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EDR-1 APPENDIX C SUPPLEMENT
SUMMARY OF REGULATORY POSITIONS TO BE ADDRESSED BY THE APPLICANT
YANKEE NUCLEAR POWER STATION YARD AREA CRANE

REGULATORY POSITION	TOPICAL REPORT SECTION		INFORMATION TO BE PROVIDED		SPECIFIC CRANE DATA
C.4.a C.4.b C.4.c C.4.d	--	1.	THE EXTENT OF ASSEMBLY CHECKOUT, TEST PROCEDURES, LOAD TESTING AND RATED LOAD MARKING OF THE CRANE.	1.	PRIOR TO INITIAL HANDLING CRITICAL LOADS, THE CRANE WILL BE GIVEN A COMPLETE ASSEMBLY CHECKOUT BY YAEC, AND THEN GIVEN A NO-LOAD TEST OF ALL MOTIONS IN ACCORDANCE WITH UPDATED PROCEDURES PROVIDED EDERER. A 125% STATIC LOAD TEST AND A 100% PERFORMANCE TEST WILL ALSO BE PERFORMED AT THIS TIME IN ACCORDANCE WITH UPDATED TEST PROCEDURES PROVIDED BY EDERER. SPACE LIMITATIONS WILL RESTRICT THE 100% PERFORMANCE TEST FROM BEING CONDUCTED FOR THE FULL RANGE OF BRIDGE MOTION, AS IT WOULD BE NECESSARY TO CARRY THE TEST LOAD OVER SPENT FUEL IN THE SPENT FUEL POOL. A NO-LOAD TEST OF ALL MOTIONS AND A TWO BLOCKING TEST WILL BE PERFORMED BY EDERER PRIOR TO DELIVERY OF THE CRANE PER TOPICAL REPORT EDR-1. THE MAXIMUM CRITICAL LOAD IS PLAINLY MARKED ON EACH SIDE OF THE CRANE.
C.5.d	III.C(C.5.a)	1.	THE EXTENT THE PROCUREMENT DOCUMENTS FOR THE CRANE'S STRUCTURE'S, WHICH WILL NOT BE REPLACED, REQUIRED THE CRANE MANUFACTURER TO PROVIDE A QUALITY ASSURANCE PROGRAM CONSISTENT WITH THE PERTINENT PROVISIONS OF REGULATORY GUIDE 1.28.	1.	NOT APPLICABLE

244 ION EXCHANGE PIT

244.1 Description

The Ion Exchange Pit is a reinforced concrete structure that contained the ion exchange vessels used to purify the Spent Fuel Pit and Main Coolant System. The Ion Exchange Pit is no longer in service, and decontamination and dismantlement have been initiated. Alternate filtering and ion exchange equipment have been installed in the Spent Fuel Pit (Section 221).

The Ion Exchange Pit shares a common wall with the Spent Fuel Pit, which will be maintained until fuel is permanently removed from the Spent Fuel Pit, at which time it can be decontaminated.

No major dismantlement activities should be performed on the Ion Exchange Pit wall common with the SFP or floor, until fuel is permanently removed from the Spent Fuel Pit.

245 PRIMARY VENT STACK

245.1 Description

The Primary Vent Stack is a steel stack that vents monitored airborne releases from the Ventilation System and the VC Ventilation and Purge System. The bottom of the stack is supported by a steel frame that is supported by the Primary Auxiliary Building.

The Primary Vent Stack is required during the dismantlement period to support decommissioning activities, and to vent air processed by both the Ventilation System and VC Ventilation and Purge System. The Primary Vent Stack must remain in service until the fuel has been permanently removed from the Spent Fuel Pit and decontamination activities in the Vapor Container, Primary Auxiliary Building, Waste Disposal Building and Spent Fuel Pit Building have been completed.

245.2 Decontamination and Dismantlement Considerations

- General decontamination and dismantlement considerations are presented in Section 200.3.3.

246 SPENT FUEL PIT AND SPENT FUEL PIT BUILDING

246.1 Description

The Spent Fuel Pit is a reinforced concrete structure that provides underwater storage of irradiated fuel and control rods and associated fuel transfer equipment (Drawing Number 9699-FM-21A). The Spent Fuel Pit inside dimensions are approximately 16 feet by 34 feet by 37 feet deep with a wall thickness that varies between 5 and 6 feet. The Spent Fuel Pit floor is constructed from a 3 foot mat located about 17 feet below grade. The Spent Fuel Pit walls and floor are lined with stainless steel to prevent leakage.

Fuel assemblies, neutron sources, canisters containing portions of the Reactor Vessel internals (core baffle and lower core support plate), and other irradiated components are stored in a two tier rack system. Currently, there are 533 fuel assemblies stored in the Spent Fuel Pit. The lower racks are an anodized aluminum support structure, to which are attached welded Boral sheets. The upper racks are modules supported on intermediate columns attached to the Spent Fuel Pit floor. The modules are comprised of boral contained between an inner and outer canister wall. Grating is installed between upper and lower racks. The racks are designed to maintain proper spacing and structural integrity after being impacted by a fuel assembly dropped onto any location from a height of six inches above the top of the racks.

The Spent Fuel Pit Building is a steel-braced frame, metal sided structure that supports the superstructure to both the New Fuel Vault and the Spent Fuel Pit. The building provides an enclosed work area and contains the Spent Fuel Manipulator Crane, the New Fuel Hoist and the SFP Cooling System pumps. Roof hatches are provided for equipment and cask access using the Yard Area Crane which is located directly above the building.

The Spent Fuel Pit will be maintained until fuel is permanently removed from the pit. While fuel is stored in the Spent Fuel Pit during dismantlement activities, special precautions will be instituted to protect the Spent Fuel Pit and associated support systems from physical damage and other adverse conditions that could occur.

The Spent Fuel Pit and Spent Fuel Pit Building must be maintained until fuel is permanently removed from the Spent Fuel Pit.

246.2 Decontamination and Dismantlement Considerations

General decontamination and dismantlement considerations are presented in Section 200.3. The following are specific area considerations:

- If fuel is stored in the Spent Fuel Pit during the dismantlement of systems and structure in proximity of the Spent Fuel Pit Building, the special precautions presented in Section 100.3 will

be instituted to restrict activities that could cause physical damage or other adverse consequences to the Spent Fuel Pit and associated support systems.

- The fuel racks and Spent Fuel Pit liner should be decontaminated prior to dismantlement.
- The Spent Fuel Pit fuel handling components (manipulator, manipulator rails, , new fuel elevator, and fuel inspection elevator) should be dismantled into manageable sections.
- The soil under the Spent Fuel Pit will be sampled after the pool has been emptied as an element of the site characterization.

247 NEW FUEL VAULT

247.1 Description

The New Fuel Vault is a reinforced concrete and concrete masonry structure. The vault is contained within a lower section of the Spent Fuel Pit Building. The west and south walls of the New Fuel Vault are common to the Spent Fuel Pit and the Ion Exchange Pit, respectively.

The New Fuel Vault is required to support plant activities until fuel is permanently removed from the Spent Fuel Pit. The common walls between the vault and the Spent Fuel Pit and Ion Exchange Pit will be maintained.

248 PRIMARY AUXILIARY BUILDING

248.1 Description

The Primary Auxiliary Building is a concrete masonry building with two stories and a partial basement at the southeast corner. All systems and components within the Primary Auxiliary Building have been dismantled and the building has been decontaminated.

249 DIESEL GENERATOR BUILDING

249.1 Description

The Diesel Generator Building contained the plant systems that supported plant response to postulated accidents and transients.

The Diesel Generator Building is no longer in service and all contaminated systems and components within the building have been dismantled, and the building decontaminated. The building has been demolitioned. Only the concrete pad/foundation, floor drains and electrical duct banks remain.

The concrete rubble generated from the demolition has been secured in an area located west of the Turbine Building.

250 WASTE DISPOSAL BUILDING

250.1 Description

The Waste Disposal Building contained systems and structures for processing, packaging, and temporarily storing waste, prior to the off-site shipment of low level radioactive waste. The structure is a steel-framed building with concrete masonry unit walls. All systems have been dismantled and the Waste Disposal Building has been decontaminated. The Waste Disposal Building shares common walls with the Warehouse, Potentially Contaminated Area (PCA) Storage Building 1, and the Compactor Building.

251 SAFE SHUTDOWN SYSTEM BUILDING

251.1 Description

The Safe Shutdown System Building contains the Fire Water Storage Tank (TK-55) Heating Boiler and associated components. The structure is constructed of reinforced concrete walls.

The Safe Shutdown System Building will be required during the dismantlement period to house the Fire Water Storage Tank Heating Boiler and to prevent the contents of TK-55 from freezing.

252 POTENTIALLY CONTAMINATED AREA STORAGE BUILDINGS 1 AND 2 AND WAREHOUSE

252.1 Description

There are three major areas located on the plant site for the storage of radioactive/hazardous materials and waste awaiting shipment:

- Potentially Contaminated Area (PCA) Storage Building 1 - PCA Storage Building 1 is used primarily for the storage of low level radioactive material prior to shipping. The structure's walls are constructed from concrete masonry units.

- PCA Storage Building 2 - PCA Storage Building 2 is used for the storage of contaminated tools and equipment. The structure's walls are constructed from uninsulated corrugated metal panels.

- PCA Warehouse - The PCA Warehouse is used for storage of low level radioactive waste, waste containers and contaminated equipment prior to shipment. The structure is a steel-framed building, with reinforced concrete masonry unit walls.

These storage areas will be used during the dismantlement period to prevent inadvertent exposure to radiation and spread of contamination. The structures will be decontaminated after all radioactive/hazardous materials stored within these areas have been permanently removed.

253 COMPACTOR BUILDING

253.1 Description

The Compactor Building contained two solid waste compactors and provides a packaging area for radioactive waste shipping containers. The structure's walls are constructed from reinforced concrete masonry units.

The building will be required during the dismantlement period to prevent inadvertent exposure to radiation and spread of contamination. The structure will be decontaminated after contaminated material processing is not required.

254 SERVICE BUILDING AND FUEL TRANSFER ENCLOSURE

254.1 Description - Service Building

The Service Building is divided into two sections. One of these sections is located in the Radiation Control Area of the plant. This section contains the primary side machine shops, control point, primary side chemistry laboratory, counting room and decontamination showers. The structure's walls are constructed from reinforced concrete masonry units.The building will be required during the dismantlement period to support dismantlement and decontamination activities. The structure will be decontaminated after most of the site decommissioning activities have been completed.

254.2 Description - Fuel Transfer Enclosure

The Fuel Transfer Enclosure (FTE) is a new structure that will serve as a work area for preparation of the NAC International (NAC) supplied fuel storage canister as part of the overall fuel loading operation.

The new FTE is a southern extension of the Service Building under the yard area crane and immediately adjacent to the Spent Fuel Building. It is a 31-ft wide by 56 ft long steel building and includes the existing North Decon Area, and the existing welding booth, which will become the security access point to the FTE. The FTE is provided with a roof hatch for Yard Crane access for the fuel storage canisters.

The building will be required during the dismantlement period to support dismantlement and decontamination activities. The structure will be decontaminated after most of the site decommissioning activities have been completed.

255 MISCELLANEOUS TANKS

255.1 Description

This section presents descriptions of plant tanks. These tanks are contaminated, potentially contaminated, or are needed to support decommissioning activities.

- Primary Water Storage Tank - The Primary Water Storage Tank (TK-39) stored demineralized water to support Spent Fuel Pit operations. The tank is a component of the Demineralized Water System (Section 227). The tank is constructed of aluminum and has an inner floating roof which restricts the absorption of air into the water. The tank may be used to store demineralized water to support decommissioning activities during the dismantlement period activities.

- Temporary Waste Water Processing Island Tanks - The Temporary Waste Water Processing Island System (Section 211) uses various tanks for operation:

  - Temporary Storage Tank - The Waste Water Storage Tank (TK-81) is a 20,000 gallon storage tank that stores contaminated liquids prior to processing. The tank is constructed of carbon steel with dimensions of 43’-5” x 11’-10” x 11’-8”. The tank is constructed so that the internal tank capacity of 20,000 gallons is bermed by a secondary tank for leak protection. This tank will be used to support waste processing during the dismantlement period.

  - Temporary Test Tanks - The two 5,000 gallon test tanks (TK-34-3 and 4) store processed liquids prior to testing and discharge from the plant. Each tank is constructed of polyethylene. These tanks will be used to support waste processing during the dismantlement period.

  - Service Building Radioactive Sump Tanks - The Radioactive Lab Waste Transfer Tank (TK-41) and the Laundry Waste Transfer Tank (TK-38) temporarily store water from the Radiation Control Area, chemistry laboratory and machine shop drains and sinks, and are connected to a common discharge line. These tanks will be maintained until Service Building decontamination activities are completed.

  - Propane Tanks – The package heating boiler and the evaporator are supplied fuel from two 1,000 gallon propane tanks (TK-79 and TK-80) located southeast of the Warehouse. These tanks will be used to support liquid waste processing during the dismantlement period.

- Fire Water Storage Tank - The Fire Water Storage Tank (TK-55) provides a back-up source of water to the Fire Protection and Detection System (Section 232). The tank is constructed of steel. The tank will be used to support operation of the Fire Protection and Detection System during the dismantlement period activities.

- Fuel Oil Storage Tanks – Description of above-ground fuel oil storage tanks is presented in Section 511.4.1.

256 METEOROLOGICAL TOWER

256.1 Description

A Meteorological Tower provides real time capability to determine wind speed and direction for on-site emergency planning purposes.

257 INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)

257.1 Description

The Independent Spent Fuel Storage Installation (ISFSI) provides dry cask storage of spent nuclear fuel and Greater-Than-Class-C (GTCC) waste generated at the YNPS. Dry cask storage of spent nuclear fuel will be conducted under the general license provisions of 10CFR72, Subpart K and 10CFR30. The YNPS will use the NAC International, Inc., Multi-Purpose Canister (MPC) system (Docket No. 72-1025), which received a certificate of compliance in April 2000.

The ISFSI is designed to store a combination of up to 18 NAC-MPC dry fuel or GTCC waste storage casks. It is located in the southwest corner of the site (Figure 257-1) and consists of a concrete pad, access road, security fences, temperature monitoring instrumentation, lighting, security, ISFSI Instrumentation Enclosure.

Spent nuclear fuel from the existing Spent Fuel Pool Building (SFPB) and GTCC waste will be transferred to the Fuel Transfer Enclosure (FTE) using a Transportable Storage Canister (TSC) in a shielded transfer cask. The FTE will serve as the work area for the preparation of the TSC before it is placed into a Vertical Concrete Cask (VCC) using the Yard Area Crane (see FSAR Section 243). The VCC, which will be loaded onto a heavy-haul trailer, and transported to the ISFSI via an access road.

257.2 Status

The ISFSI construction is complete. Spent nuclear fuel transfer is scheduled to begin in 2001.

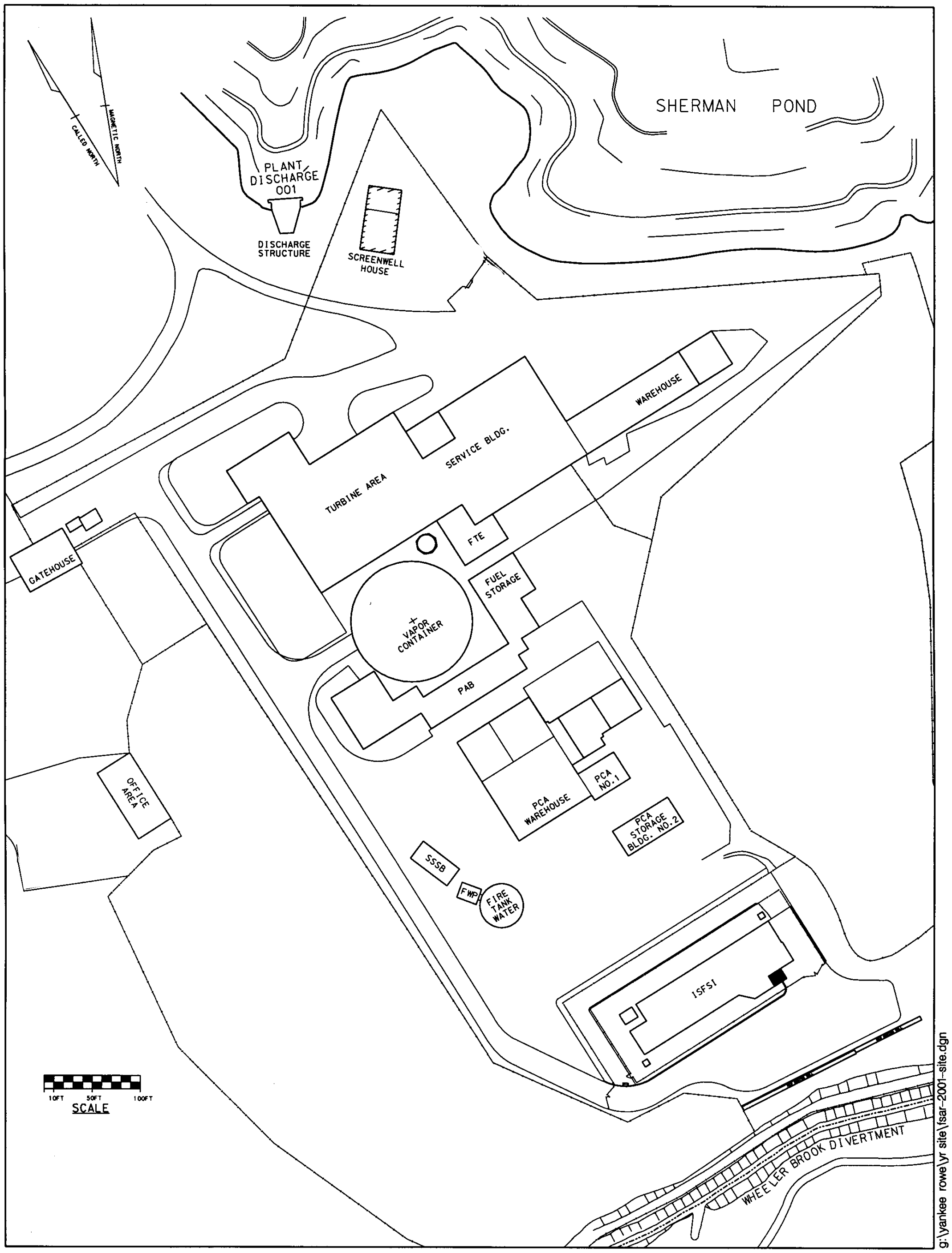

FIGURE 257-1

2001 Site Design

300 ENVIRONMENTAL – SITE CHARACTERISTICS

300.1 Demography

YNPS is located in the Berkshire Hills of Franklin County in Rowe, Massachusetts. The site is at the bottom of a deep valley along the Deerfield River on the southeast bank of Sherman Reservoir. The area surrounding the site is mostly wooded with very steep gradients on both sides of the Deerfield River.

The YNPS site boundary and plant exclusion area is shown in Figure 300-1.

300.2 Geography and Land Use

The population density in the rural area surrounding the YNPS site is low. Within one mile of the site, for example, the population is 48 (based on 1989 Massachusetts municipal census counts). The populations of the two closest towns, Rowe and Monroe, are 354 and 141, respectively. The nearest population center of 25,000 or more is Pittsfield, Massachusetts, located about 21 miles south-west of the site. The regional population is expected to remain virtually unchanged over the next decade.

Land use near the site is made up of a few farms and some commercial businesses. The centers of Rowe and Monroe have small, mixed clusters of local businesses, municipal buildings, and residences, with homes scattered throughout the area. There is no large industrial activity within five miles of the site; the only industry in the area is the YNPS and several hydroelectric facilities along the Deerfield River.

The nearest highway and railroad are each about five miles south of the site. The closest airport is in North Adams, Massachusetts, about ten miles west of the site.

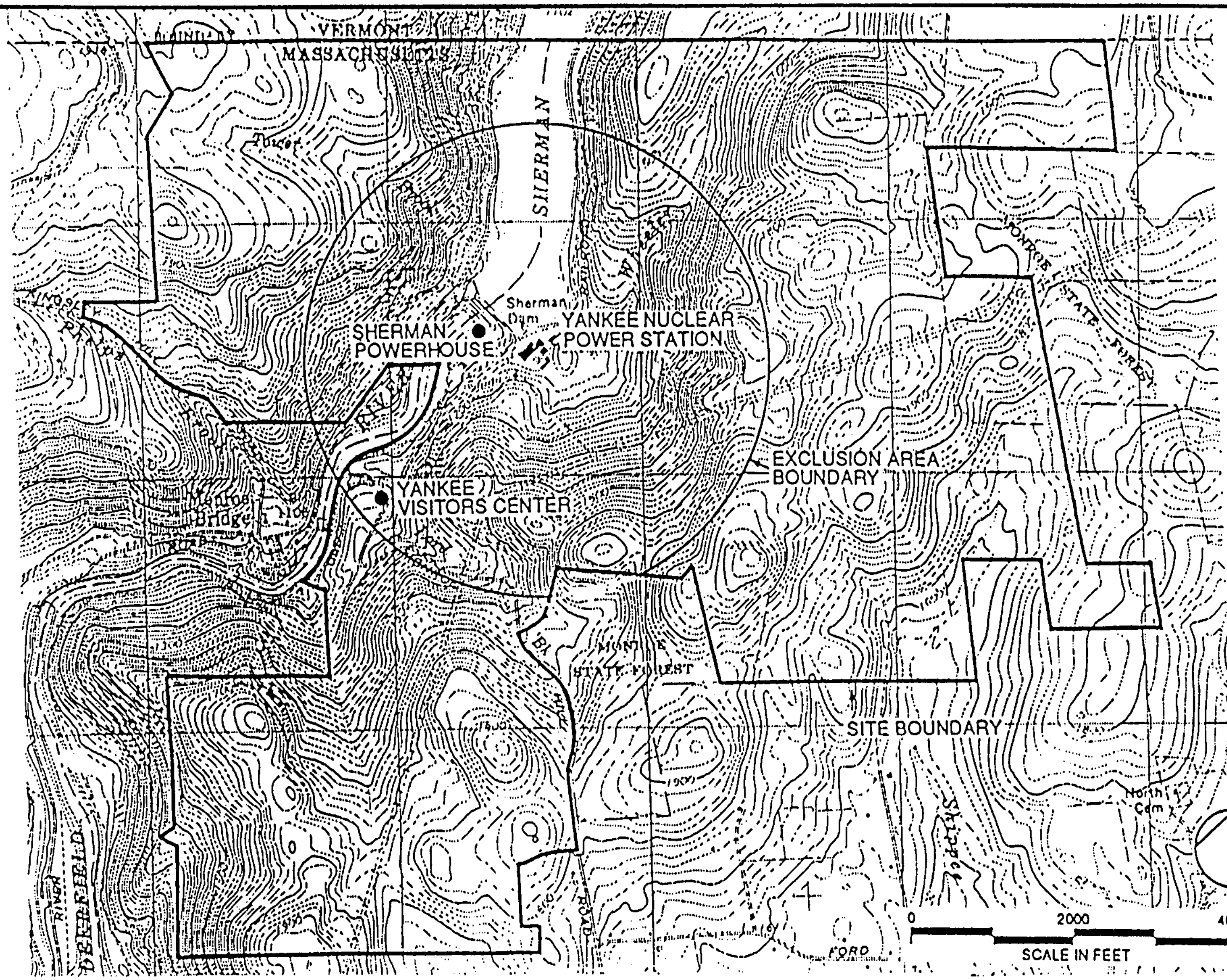

FIGURE 300-1

Site Boundary and Plant Exclusion Area

301 METEOROLOGY (GENERAL CLIMATE, SEVERE WEATHER)

The site lies in the prevailing westerlies, the belt of eastward moving air that is found in middle latitudes. Many storms pass over Massachusetts compared to other parts of the United States. This is a result of extensive air masses originating at higher and lower latitudes. The three major air mass types that affect the site are cold, dry, subartic air from Canada; warm, moist air from the Gulf of Mexico; and cool, damp air from the North Atlantic Ocean.

The hills on either side of the site rise about 1000 feet above ground level within one mile and extend from 12 miles north to 8 miles south southeast of the site. This feature affects the winds. There is, for instance, a high frequency of occurrence of “channel flow” up and down the Deerfield River valley. Also, night -time drainage flow down the east side of the river valley occurs frequently.

Normal daily temperatures range from 10°F in January to 80°F in July. Recorded extreme temperatures for the site region are -25°F and 98°F. Thunderstorms occur about 28 days per year; the annual flash density of ground lightning strikes is four flashes per square kilometer.

The site design wind speed (defined as the “fastest-mile” wind speed at 30 feet above the ground with a 100 year return period) is 80 mph. Hail storms occur about two days annually and freezing rain about 12 days per year. The maximum radial ice thickness expected for the region is 1.25 inches. Mean annual snowfall at the site totals 100 inches, with maximum snow depth on the ground totaling about 40 inches.

302 HYDROLOGY

The site is located on the east bank of the Deerfield River adjacent to Sherman Reservoir, which serves as the source of the cooling water for the plant. The drainage area upstream of the plant is characterized by a dendritic pattern, is 236 square miles, and has an average annual rainfall of between 40 and 50 inches.

The Deerfield River flow is highly regulated by two large, upstream hydroelectric reservoirs. The average long-term flow near the plant is 738 ft^3/sec.

Bedrock in the region is not a significant source of groundwater nor are there major bedrock aquifers within the site area. The direction of groundwater flow under the site is from the recharge areas on the slopes surrounding the plant toward the Deerfield River.

303 GEOLOGY AND SEISMOLOGY

The regional bedrock geology is complex; bedrock age ranges from 100 million to over one billion years old and is comprised of mostly a mosaic of metamorphic and igneous rocks. The youngest deposits are glacial soils, 10 to 12 thousand years old. Most volcanic and sedimentary rock in the area are now metamorphosed.

The site is situated on dense Wisconsinian-aged glacial till. YNPS structures are founded on this till, which ranges from 0 to 140 feet thick across the site. Bedrock under the till is part of the lower Cambrian Hoosac formation and consists of quartz-albite-biotite gneiss and a rusty gneiss in adjacent areas. Underlying these are garnet schist and a layered gneiss with some dolomitic marble, with the latter units belonging to the lower Cambrian or older Cavendish formation. A south-plunging anti-cline, whose axis is just east of the site, defines local bedrock structure.

Site bedrock is hard, internally welded metamorphic rock, not subject to significant deterioration. Bedrock fracturing is not a prominent structural feature of this bedrock; outcrops exhibit either no joints or minor, discontinuous joint surfaces. Fracture pattern analysis of site vicinity joints, joint sets, and faults show no anomalous trends for fractures. This suggests the absence of any through-going zones of post-metamorphic faulting or shear.

The site is in the Western New England Fold Belt province. It borders the Adirondack Uplift province to the west, the Valley and Ridge province to the southwest, and the New York Recess to the south. The Southeast New England Platform and Merrimack Synclinorium occur to the east and northeast, respectively.

Regional seismic events are very infrequent and do not cause surface faulting. Only two events in the province were greater than Intensity V (MM). These events were 125 miles and 210 miles from the site. The site seismic design level for new seismically qualified installations at the plant is a peak ground acceleration of 0.19 g. The return period is between 10,000 and 100,000 years.

304 PLANT WATER SUPPLY

Cooling water supply for the plant comes from the Sherman Reservoir, adjacent to the site. Potable water is supplied by an on site bedrock well.

305 PLANT EFFLUENT

The same liquid effluent release path will be used during decommissioning as was used during plant operation. All liquid discharges will be controlled in accordance with the National Pollutant Discharge Elimination System (NPDES) permit. Dilution water will be taken from the Sherman Reservoir via the Auxiliary Service Water System.

Surface water use downstream from the site is mostly for recreation and sport fishing, with limited irrigation. Water supply for the municipalities within five miles downstream of the plant is from private wells. The closest public water supplies are well fields 20 to 25 miles downstream of the plant.

306 ENVIRONMENTAL RADIOLOGICAL STATUS

306.1 Facility Operating History

Yankee Nuclear Power Station (YNPS) achieved initial criticality in 1960 and began commercial operation in 1961. On October 1, 1991, YNPS shut down after operating 31 years. On February 26, 1992, the YAEC Board of Directors decided to cease power operation permanently. YNPS operated with an average lifetime capacity factor of about 74%.

There were occurrences, during the operation of YNPS, that resulted in contamination of structures and components inside buildings located in the Radiation Control Area. In addition, several occurrences resulted in the contamination of the grounds outside of the buildings but inside the Radiation Control Area. Most of these events were minor, resulting in minimal contamination. Following detection of an event, actions were taken to remove and control contamination and to institute corrective actions to preclude future occurrences.

Table 306.1 presents a summary of significant radiological contamination occurrences at YNPS. The primary sources used to compile and review events were: Control Room Logs, Abnormal Occurrence Reports, Licensee Event Reports, Plant Information Reports and reports to the Atomic Energy Commission, Nuclear Regulatory Commission and the Commonwealth of Massachusetts. Interviews were also conducted with present employees, recently discharged employees and retired long-term employees. The purpose of the interviews was to gain additional information concerning plant occurrences and operations that may have resulted in residual contamination.

The most significant contamination event at YNPS was leakage from the Ion Exchange Pit. Significant leakage was first identified in May 1964. An unsuccessful attempt to isolate the leak was made in July 1964 by installing and sealing a concrete plug in the pit sump. Following that attempt, a decision was made to empty the resin capsules and drain the pit. This activity was completed in April 1965. In May 1965, a crack in a vertical joint at the northwest corner of the Ion Exchange Pit was found after draining the pit. The crack was repaired and the Ion Exchange Pit floor and walls were sealed to prevent further leakage.

In 1965, tritium was detected in Sherman Spring. The presence of tritium was attributed to migration of tritium from the Ion Exchange Pit into the groundwater. At the present time, the tritium concentration is significantly below the Environmental Protection Agency community water system limit (Section 306.3 and Reference 306-11). In addition, the water from Sherman Spring, as well as the Deerfield River, into which it flows, is not used for human consumption.

Systems in contact with the main coolant have been contaminated with activated corrosion products and fuel residue. In 1977, YNPS began converting from stainless steel clad fuel to zircaloy clad fuel. Following conversion, fuel clad failures began occurring. Most of the failures were minor failures of clad integrity, releasing iodines and other fission gasses into the main coolant. However, after Cycle 13, a number of peripheral fuel rods failed as a result of damage caused by water jetting from the core baffle spacer plates. These fuel failures were sufficient to cause contamination of the Reactor Vessel and Main Coolant System with fuel residue. Baffle jetting damage was eliminated by the addition of spacer plugs at the bottom of the core baffle spacer plates and modifying fuel assembly design. No significant fuel failures occurred after the modifications were implemented.

Primary system integrity was very good during plant operations. Minor leakage from Main Coolant System flanges and valve stems to the Vapor Container occurred during plant operations. However, leakage rates were maintained below Technical Specification limits. The Vapor Container was decontaminated during refueling outages to remove contamination that may have accumulated during previous operating cycles. The YNPS steam generators performed well during the 31 year operating life. There was no significant leakage between the primary and secondary sides of the generators.

The results of the historical review were incorporated into the design and implementation of the radiological scoping survey and sampling program.

306.2 Radiological Scoping Survey

The Radiological survey methodology will be discussed in YAEC's future submittal of a License Termination Plan.

306.3 Radiological Environmental Monitoring Program

The Yankee Decommissioning Quality Assurance Program establishes the Radiological Environmental Monitoring Program. The purpose of the program is to monitor the radiation and radionuclides in the environs of the plant. The program includes the following elements:

- Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters presented in the YNPS Off-Site Dose Calculation Manual.

- Performance of a land use census to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the monitoring program are made if required.

- Participation in an interlaboratory comparison program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the Quality Assurance Program for environmental monitoring.

Technical Specification 6.8.2 requires the submittal of an annual report to the NRC. The report shall include summaries, interpretations and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period.

The Radiological Environmental Monitoring Program was initiated in 1958, about two years before YNPS began commercial operation. The program has been executed continuously since its inception. In recent years, the program has incorporated continuous monitoring of air and automatic composite sampling of river water, as well as sampling of groundwater, river sediment, fish, cow milk, goat milk, vegetables, and maple syrup. Direct radiation exposure measurements, both in the immediate plant vicinity and at more distant locations, have been made continuously through the use of a network of thermoluminescent dosimeters (TLDs).

The Radiological Environmental Monitoring Program will continue to be updated and modified to reflect changes to the potential source term, the plant environs, and the surrounding land use during YNPS decommissioning.

306.4 Radiologically Affected Area Identification

- Impacted Areas

  Impacted areas have a potential for radioactive contamination (based on historical data) or contain known radioactive contamination (based on past or preliminary radiological surveillance). This includes areas where 1) radioactive materials were used and stored; 2) records indicate spills, discharges or other unusual occurrences that could result in the spread of contamination; and 3) radioactive materials were buried or disposed. Areas immediately surrounding or adjacent to these locations are included in this classification because of the potential for inadvertent spread of contamination.

- Non-Impacted Areas

  Non-impacted areas are identified through knowledge of site history or previous survey information where there is no reasonable possibility for residual radioactive contamination. The criteria used for this segregation need not be as strict as those used to demonstrate final compliance with the regulations. However, the reasoning for classifying an area as non-impacted should be maintained as a written record. An impacted area's classification may change as the radiation survey and site investigation process progresses.

306.5 Site Characterization Surveys

Site characterization is the next phase in the radiological survey process. The purpose of the site characterization surveys is to define more precisely the extent and magnitude of the contamination on site. Site characterization surveys will be used to supplement the scoping survey data in areas where data are missing or where the data indicate contamination levels are at or near the release criteria. The level of effort is related to the data needs for the item or area being surveyed. Items or areas that are highly contaminated require less data than items or areas that are near the release criteria to make decisions regarding their radiological status. Scoping and characterization data will be used to plan and complete decommissioning activities.

306.5.1 Program Description

The site characterization survey will collect additional radiological data and samples in areas to facilitate decommissioning planning. The following items will be considered in the site characterization survey for systems, structures, and components:

- Measure the activity and contamination levels associated with inaccessible plant areas before removal activities.

- Perform additional testing on contaminated concrete to improve estimates of contamination levels and depth of contamination.

- A detailed radiological survey of the Reactor Vessel, which was performed following completion of the Component Removal Project.

- Determine the extent of contamination of nonradiological plant systems, structures, and components.

- Determine the extent of activated structures and components external to the Reactor Vessel.

The following items will be considered in the site characterization survey for soil and groundwater:

- Perform additional surface and subsurface soil sampling as well as in situ measurements to determine the lateral and vertical extent of contamination identified during the radiological scoping surveys. The sampling should include areas identified with significant soil contamination as well as soil under buildings to the extent practicable.

- Establish the expected cesium-137 background concentration for the plant site. This determination should include analysis of additional off-site samples of varying soil types.

- Develop a computer model to define the groundwater regime for the site. Additional wells may be installed to provide the database needed to perform the modeling calculations.

306.5.2 Implementation Schedule

Site characterization surveys of YNPS systems, structures, and components will be completed to support detailed planning activities associated with their decontamination and dismantlement. The surveys for soil and groundwater will be completed to support detailed planning activities associated with preparation of the site for the final radiation survey. These surveys will be maintained on-site and available for inspection.

REFERENCES

306-1 NUREG/CR-5849 (Draft), “Manual For Conducting Radiological Surveys in Support of

306-2 YRC-1024, “Basis for the Radiological Status of Plant Systems and Structures,” P. Hollenbeck, September 1993.

306-3 REG 90/93, “Dose Conversion Factors For Yankee Rowe Piping,” Y. J. Yu to R. A. Mellor, April 2, 1993.

306-4 YRC-1031, “Yankee Rowe Component Activation Analysis,” K. J. Morrissey, September 1993.

306-5 NUREG/CR-3474, “Long-Lived Activation Products in Reactor Materials,” August 1984.

306-6 REG 147/93, “Dose Rate Estimate of the YNPS Reactor Vessel,” Y. J. Yu to R. A. Mellor, June 4, 1993.

306-7 “YNPS Outdoor Site Scoping Data,” F. X. Bellini, E. R. Cumming to P. Hollenbeck, September 24, 1993.

306-8 YNPS Off-Site Dose Calculation Manual.

306-9 NUREG/CR-5512, Vol. 1, “Residual Radioactive Contamination from Decommissioning - Technical Basis for Translating Contamination Levels to Annual Total Effective Dose Equivalent,” Final Report, October 1992.

306-10 1992 Annual Radiological Environmental Operating Report.

306-11 BYR 94-047, RAI on YNPS Decommissioning Plan and Decommissioning Environmental Report.

306-12 NUREG-1575, “Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM),” December 1997.

306-13 Letter, D. K. Davis (YAEC) to NRC, “Withdrawal of Proposed License Amendment to Approve Yankee Nuclear Power Station’s License Termination Plan,” May 25, 1999.

TABLE 306.1

Summary of Significant Radiological Contamination Events

Date	Description of Occurrence	Radiological Consequences
9/20/61 Incident Report No. 61-15	A main coolant sample container was dropped between the Primary Auxiliary Building and the Waste Disposal Building. The contents of the container spilled onto the asphalt surface.	About 35 μCi of radioactivity was released as a result of the spill on the asphalt surface. Fixed contamination remaining following decontamination resulted in a contact dose of about 0.05 mR/hr.
9/18/63 Incident Report No. 63-12	A sampling valve located over the Ion Exchange Pit was inadvertently left open when filling the Shield Tank Cavity from the Safety Injection Tank. Water from the tank spilled onto the cover of the Ion Exchange Pit. A portion of the water drained from the cover onto the asphalt on the west side of the Ion Exchange Pit.	Radiation levels of the spill area were 70-100 mR/hr measured 1 inch from the surface. Ion Exchange Pit cover contamination levels were 1,000,000-10,000,000 dpm measured over several square inches. Asphalt contamination levels were 20,000-60,000 dpm/ft^2. Both the Ion Exchange Pit cover and the asphalt were decontaminated.
10/8/63 Incident Report No. 63-17	Several small holes were detected on the bottom of a leakage collection drum located in a storm drain catch basin outside of the Spent Fuel Building. The drum was used to collect leakage from the Fuel Transfer Chute Dewatering Pump.	Water with a radioactivity level of about 6E-5 μCi/ml leaked from the drum into the catch basin. The basin was pumped to the Waste Disposal System and flushed with Service Water.

Date	Description of Occurrence	Radiological Consequences
9/3/64 Incident Report No. 64-8	A spill of main coolant occurred while filling and pressurizing the shutdown cooling pump gland seal water tank. Main coolant flowed out the vent connection into a relief valve discharge header and onto the Primary Auxiliary Building roof.	About 270 µCi of radioactivity was released as a result of the spill. Several samples were taken: Seal tank drain 2E-3 µCi/ml PAB roof 1E-3 µCi/ml Storm drain1 E-6 µCi/ml Predominant isotopes were Co-58, Co-60 and Mn-54. The area was decontaminated. Several years after the event, a new roof was installed and the old roofing material was disposed of as radioactive material.
10/3/64 AOR No. 64-13	A valve was inadvertently left open after adding water to the Ion Exchange Pit. Water continued to flow into the Ion Exchange Pit from the Primary Water Storage Tank. Water leaked from the Ion Exchange Pit into the ground resulting in water coming up through the asphalt on the west side of the pit.	Leakage from the Ion Exchange Pit contaminated the ground and asphalt on the west side of the Ion Exchange Pit. The water had a specific activity of 8E-8 µCi/ml and was below the maximum permissible concentration. The area was flushed with service water to dilute the contamination as it entered the storm sewer.
2/17/65 AOR No. 65-6	A routine chemistry sample of Ion Exchange Pit water indicated an activity level of 1.2E-5 µCi/ml. This was about 100 times above the normal level. The cause was identified as a leaking outlet connection on a cation bed that was in main coolant service. The activity decreased by about 50% within a day after isolating the leak. The connection was installed 19 days before the leak was identified.	Analysis indicated that the activity was comprised primarily of F-18 (half life = 1.8 hr) and an unknown isotope with a gamma energy of 0.14 MeV (half life = 40 hr). Based on an existing leak from the Ion Exchange Pit to the soil, about 3,400 µCi in 74,000 gallons of water was released to the area beneath the Ion Exchange Pit over the 19-day period.

Date	Description of Occurrence	Radiological Consequences
3/25/65 AOR No. 65-8	A sample of Ion Exchange Pit water indicated a high activity level, 2E-4 μCi/ml. The cause was identified as a leak from an anion exchange capsule and filter that were installed the previous day. Main Coolant System purification flow was stopped and all capsules were isolated.	Main coolant leaked into the Ion Exchange Pit, increasing the radioactivity level. After the leak was isolated, a feed and bleed dilution of the Ion Exchange Pit was established. About 88,000 gallons of demineralized water was added to the Ion Exchange Pit.
5/64 – 5/65 Operation Reports Nos. 42-53	Significant Ion Exchange Pit leakage was identified in May 1964. An unsuccessful attempt to isolate the leak was made in July 1964 by installing and sealing a concrete plug in the pit sump. Following that attempt, a decision was made to empty the resin capsules and to drain the pit. This activity was completed in April 1965. In May 1965, a crack in a vertical joint at the northwest corner of the Ion Exchange Pit was found after draining the pit. The crack was repaired and the Ion Exchange Pit floor and walls were sealed to prevent further leakage. In 1965, tritium was detected in Sherman Spring. The presence of tritium was attributed to migration of tritium from the Ion Exchange Pit into the groundwater. Section 3.1.5 presents additional information regarding the activity in Sherman Spring.	Water from the Ion Exchange Pit leaked into the soil below the pit during the period between May 1964 and April 1965. The average activity during this period was 7E-6 μCi/ml resulting in a total release of about 36,000 μCi of identifiable activity. Greater than 95% of this activity was due to Cr-51, Mn-54, Fe-59, Co-58, Co-60, Ag-110m, and Hf-181. Based on the average activity and the lowest maximum permissible concentration, the release from the Ion Exchange Pit was less than 12% of the maximum permissible concentration for unrestricted areas. Based on soil characteristics it is unlikely that this activity migrated to an unrestricted area. The amount of tritium attributed to this leakage has been estimated to be less than 200 Ci (Reference 306-11).
4/8/66 AOR No. 66-3	Main coolant was inadvertently aspirated into the Primary Vent Stack fan from the Low Pressure Vent Header during outage related Reactor Vessel venting operations. Main coolant leaked from the riveted joints in the fan duct work onto the Primary Auxiliary Building Mechanical Equipment Room floor.	Main coolant was spilled with a radioactivity level of 2.26 μCi/ml tritium and 3.4E-2 μCi/ml gross beta-gamma. Smears of the affected area indicated 100,000-1,000,000 dpm/ft^2. A survey of the vent line and the duct work indicated contact radiation levels of 10-70 mR/hr on horizontal runs and 10-25 mR/hr on vertical runs. The area was decontaminated.

Date	Description of Occurrence	Radiological Consequences
9/27/66 AOR No. 66-7	The Spent Fuel Pit overflowed when a Primary Water Storage Tank valve was inadvertently left open. Most of the water flowed from the building, down the north exterior wall, over a small section of asphalt paving and into a storm drain. A few gallons leaked into the New Fuel Vault.	A sample of the water that overflowed from the Spent Fuel Pit indicated 3.2E-5 µCi/ml gross activity and 5.4E-3 µCi/ml tritium. This occurrence resulted in a total release of 4 µCi gross beta-gamma activity and 670 µCi of tritium. Affected areas were decontaminated, the storm drain was flushed with about 75,000 gallons of service water.
9/27/66 AOR No. 66-8	A sample of the west side storm drain culverts indicated elevated activity levels. This culvert was not affected by the Spent Fuel Pit spill. The source of activity was traced to a leak from the Safety Injection Tank heating system safety valve located in the Primary Auxiliary Building. Floor drains in this section of the building discharge to a storm drain on the south side of the building which discharge into the west culvert.	Samples collected from the west side culvert indicated 6.7E-7 µCi/ml average gross beta-gamma activity. The Safety Injection Tank water analysis indicated 3E-5 µCi/ml gross beta-gamma activity and 1.1E-1 µCi/ml tritium. About 0.8 µCi gross beta-gamma activity and 3.32 millicuries of tritium were released into the west culvert. Affected surface areas were decontaminated.
11/1/66 AOR No. 66-9	A temporary hose failed during a routine drainage operation on the Fuel Transfer Chute pump discharge line. The spilled water drained into a storm drain served by the east culvert.	The spilled liquid had an activity of 3E-3 µCi/ml, resulting in a release of about 113 µCi into the storm drain. The storm drain was flushed with about 250 gallons of water.
2/18/72 AOR No. 72-3	Moisture was detected under insulation on the Test Tank level indicator. A leak developed from the indicator to the ground at the base of the tank coincident with the discovery. The leak was contained within 15 minutes of discovery.	The leakage from the tank had a radioactivity content of 0.0004 µCi beta-gamma and 2,017 µCi tritium. About 4 ft^3 of crushed stone and soil at the base of the tank was removed for disposal. Analysis of the soil following removal indicated no radioactivity above background levels.

Date	Description of Occurrence	Radiological Consequences
7/16/75 PIR No. 75-7	While walking near the Ion Exchange Pit a technician received an alarm on a portable radiation detection monitor. Upon investigation, an area of bare soil with radioactive contamination of about 500,000 dpm was detected. Contamination appeared to be limited to several square feet of soil with most of the activity limited to an area of about 1 ft^2. A complete surface survey within the plant protected area fence was initiated using RM-14 radiation detectors with HP-210 probes held approximately 1 inch off the ground. Fourteen additional areas of contamination were identified. Ten of the fourteen areas were located on the nonradiological side of the plant.	Several sources of contamination were identified: 1) single specks of radioactive material, 2) soil contaminated with relatively low activity to a depth of several inches, and 3) pieces of contaminated materials (e.g., wood, concrete, polyethylene) on top of the ground. Low levels of radioactivity were also detected in the normal rain water runoff paths. Analyses of several soil samples indicated that the radionuclides consisted of nuclides with half-lives greater than one year (predominately Co-60). The contamination most likely was deposited several years prior to detection. Contaminated soil and debris were removed, including soil at the bottom of the storm drain catch basins. Any remaining soil contaminated with residual, low level radioactivity was sprayed with an asphalt sealer and covered with a thin layer of soil. Paved areas were swept and sealed with an asphalt sealer.
8/10/77 PIR No. 77-10	Routine analysis of the laboratory demineralized water supply indicated the presence of tritium. The cause was inferred to be a valve line-up error that allowed main coolant to flow into the demineralized water header.	Contaminated water was detected in the Demineralized Water Storage Tank and the Auxiliary Boiler Condensate Tank. The estimated activity released to the secondary plant was 4,000 μCi tritium and 75 μCi nonvolatile activity. The Demineralized Water Storage Tank was discharged as a permitted release and refilled with clean water. Normal boiler drum blowdown reduced the radioactivity to undetectable levels.

Date	Description of Occurrence	Radiological Consequences
12/21/77 PIR No. 77-16	A pipe from the Service Building sump to the Gravity Drain Tank was severed during core boring activities. The sump tanks receive water from the Radiation Control Area sinks, chemistry laboratory, and machine shop drains. A mechanical float-switch assembly starts and stops the pump which transfers liquid from the sump tank to the gravity drain tank. Water was released from the system before the pump could be secured.	The contaminated water released the following radioactivity into the soil: I-131 16.50 µCi I-133 2.76 µCi Cs-134 0.34 µCi Cs-137 0.50 µCi Co-60 0.58 µCi The soil at the location of the rupture was removed and disposed of.
10/24/79 PIR No. 79-4	Routine analysis of the auxiliary plant systems indicated the presence of tritium. The cause was inferred to be valve leakage between the Test Tank transfer pumps and the Primary Water Storage Tank. Water was transferred to the Auxiliary Boiler Condensate Tank, and the Demineralized Water Storage Tank during normal make-up operations.	The radioactivity level of water in the demineralized water header was very low: 5E-5 - 3E-4 µCi/ml tritium and 1E-8 - 1E-7 µCi/ml Cs-134, Cs-137, Co-60, Mn-54 activity.
8/6/80 PIR No. 80-9	Several gallons of contaminated water and about one quart of resin were released through a pinhole leak in a hose while transferring spent resin from the Ion Exchange Pit into a shipping cask.	Resin contact radiation levels were less than 1 mR/hr and spilled liquid readings were about 300,000 dpm/100 cm^2. The spill area was decontaminated to remove contamination, including excavation of some asphalt.
5/15/81 PIR No. 81-9	The reactor vessel head was bumped against the side of the equipment hatch during removal from the Vapor Container. Contaminated material from the underside of the head was released and fell to the asphalt below the equipment hatch.	General contamination levels on the asphalt below the equipment hatch were 1,000-500,000 dpm/100 cm^2. An area about 30 ft by 50 ft was contaminated with a total activity of about 250 µCi. About 10 µCi was discharged to Sherman Pond when rain washed the radioactive material into the east storm drain before the area could be decontaminated.

Date	Description of Occurrence	Radiological Consequences
9/10/84 PIR No. 84-16	A failed PVC drain line from the Waste Disposal Building was discovered during soil excavation activities. The drain line had six pipe joints, each of which leaked apparently due to failure of the solvent welds.	Soil below one of the pipe joints was significantly contaminated. Analysis indicated 50,000 dpm over the area of maximum contamination. One “hot spot” contained 29,300 pCi/g Co-60. Average Co-60 activity, measured 2 feet below the pipe joint was about 2,100 pCi/g. Average Cs-137 activity at this location was about 17 times less than the average Co-60 activity. Soil under all pipe joints was removed to a depth of 5-7 feet below plant grade. About 420 cubic feet of soil and rock were removed and disposed of as radioactive waste. All areas above the excavation were sealed under a concrete cap.
12/14/91 PIR No. 91-7	Analysis of the laboratory demineralized water supply, the Demineralized Water Storage Tank, and the Auxiliary Boiler Condensate Tank indicated the presence of tritium and boron. No activity was found in the Primary Water Storage Tank. The root cause of the event could not be determined.	About 840 μCi of tritium was leaked into the demineralized water supply. Water was discharged from the Demineralized Water Storage Tank as a permitted release and the demineralized water header was flushed.

307 FINAL RADIOLOGICAL SURVEY – SITE RELEASE CRITERIA

This information has been extracted, updated and relocated to the PSDAR within the FSAR.

400 TRANSIENTS

400.1 General Overview

This section presents an accident analysis that assesses the impact of decommissioning on both occupational and public health and safety. A structured, comprehensive process was used to identify and evaluate events that could occur during the period from approval of the Decommissioning Plan through completion of the final radiation surveys (Reference 400-1). The accident analysis considered: (1) decommissioning activity events, (2) loss of support system events, (3) fire events, (4) explosion events, (5) external events and (6) spent fuel storage events. Analysis of decommissioning events included all phases of decommissioning activities: decontamination, dismantlement, packaging, storage and radioactive materials handling.

The risk of accidents resulting in a significant radiological release during decommissioning activities is considerably less than that during plant operations. YAEC evaluated all of the Final Safety Analysis Report Section 400 safety analyses for applicability to a permanently defueled condition (Reference 400-2). The only design basis event remaining applicable was the Spent Fuel Pit fuel handling accident. The remaining events which could occur in a permanently defueled condition and impact the health and safety of the public are related to the release of airborne radioactive materials during decommissioning activities.

400.1.1 Radionuclide Release Limits

Prior to the decision to permanently cease power operations, radiological releases resulting from design basis accidents postulated in evaluations and the Final Safety Analysis Report were evaluated using dose reference values from 10CFR Part 100. The 10CFR Part 100 reference values limit dose to an individual at the Exclusion Area Boundary during the first two hours following the onset of a postulated fission product release. The dose limits are less than 25 rem total dose to the whole body and less than 300 rem thyroid dose from radioactive iodine.

Since the decision to permanently cease power operations, YAEC requested and received an exemption from the emergency preparedness requirements of 10CFR50.54(q) (Reference 400-3). This exemption allowed YAEC to discontinue off-site emergency response activities and to refocus the scope of on-site response capability. NRC approval of the exemption was predicated on the absence of accidents at a level of severity where the off-site dose could exceed the Environmental Protection Agency (EPA) Protective Action Guides (PAGs). Off-site protective actions are not warranted if the off-site dose following a postulated accident is less than the EPA PAGs.

The EPA PAGs are limiting values based on the sum of the effective dose equivalent resulting from exposure to external sources and from the committed effective dose equivalent incurred from all significant inhalation pathways during the early phase of an event (Reference 400-4):

	EPA PAGs, rem
Whole Body	1
Thyroid	5
Skin	50

Releases resulting from accidents postulated in the decommissioning accident analysis were evaluated using the EPA PAGs as an upper limit. This ensures that the current Defueled Emergency Plan remains adequate for decommissioning and eliminates the need to reinstitute an off-site emergency response capability. Use of the EPA PAGs as an administrative limit also ensures that postulated accident off-site doses are significantly less than the 10CFR Part 100 reference values.

400.1.2 Assumptions

The following assumptions have been incorporated into the accident analysis:

- Special Nuclear Material used as reactor fuel will not be moved into the Reactor Vessel. This a condition of the YNPS possession only license.

- The airborne pathway is the dominant radioactivity release pathway. Activities that could result in release of radioactive liquids will be designed to contain the releases within the liquid waste processing system using existing or supplemental barriers.

- Airborne releases are assumed to occur at ground level with a conservative dispersion factor of 2.84E-04 sec/m^3 (Reference 400-5).

- Direct failures and consequences of initiating events were considered in the consequence analyses. Separate, coincident, random failures were not considered.

- Decommissioning activities are independent from each other. There are no credible common cause mechanisms that could result in the simultaneous release of radioactivity from multiple activities that would exceed the equivalent release of the radioactive contents of the single, bounding container or component that results in the highest off-site dose. Interactions between systems during radioactive materials handling activities will be precluded by the maintenance of safe load paths; protective zones around limiting systems, structures and components; and single handling criteria for higher contamination items. The consequences of fire and explosions may impact several activities simultaneously. These events are considered separately in Sections 405 and 406.

- Decommissioning activities may be performed in the Vapor Container without isolating the Vapor Container from the environment. However, the capability to isolate the Vapor Container will be retained to mitigate the consequences of a significant radioactive release. If an accident occurs, the Vapor Container will be isolated expeditiously. Vapor Container isolation is the closure of all penetrations and openings to restrict transport of airborne radioactivity from the Vapor Container atmosphere to the environment. Pressure retention capability is not necessary.

400.2 Event Identification Process

A structured, comprehensive process was used to identify accident initiating events which could lead to radionuclide releases during the YNPS decommissioning (Reference 400-1). The process

included development of a logic diagram to evaluate all phases of decontamination, dismantlement and fuel management activities, as well as to identify nonradiological events.

Accident initiating events were grouped by structures, systems and components within a plant area. The accident initiating events were compared using previous YNPS accident analyses, as well as current evaluations and calculations to identify the dominant accident initiating events within each plant area and then among plant areas. Accident scenarios were developed for these dominant accident initiating events. The scenarios formed the bases and inputs for radiological dose calculations to determine the impact on the health and safety of the public.

The following events were considered in the accident analysis and are presented below:

- Events affecting occupational health and safety, including radiological and nonradiological events.
- Off-site events affecting public health and safety.
- Nonradiological events affecting public health and safety.
- Radiological events affecting public health and safety, including the following:
    - Decommissioning activity events, including decontamination, dismantlement, packaging, storage and materials handling.
    - Loss of support system events, including loss of off-site power, cooling water and compressed air.
    - Fire and explosion events.
    - External events.
    - Spent fuel storage events, including fuel handling event, loss of spent fuel cooling capability, and interactions between spent fuel and decommissioning activities.

400.3 Summary

The accident analysis assessed the impact of decommissioning on both occupational and public health and safety by considering decommissioning and fuel storage events. The evaluation of events that could affect occupational health and safety indicated that implementation of the Radiation Protection, Occupational Safety Programs and Defueled Emergency Plan ensures that these events are sufficiently minimized. Analysis of events that could affect public health and safety indicated that there were no events that could significantly affect public health and safety.

Table 400.1 provides dose conversion factors for several radionuclide mixes (Reference 400-8). The dose conversion factors are in units of rem per curie released to the atmosphere. They apply at the Exclusion Area Boundary and were determined as of January 1994. Table 400.2 provides a summary of the radioactivity content of several systems as of the same January, 1994 date.

Due to radioactive decay, the dose conversion factors for representative radionuclide mixes will change with time. A bounding set of dose conversion factors has been calculated that will be applicable until the date 1/1/2005 (Reference 400-19). These were determined by decaying several radionuclide mixes until that date and using the most limiting result.

The analyses identified requirements that must be implemented during decommissioning to ensure that the accident analysis basis is maintained. Table 400.3 summarizes these items. The analyses also present planning considerations that reduce the probability of occurrence or consequences of the events that were evaluated. Table 400.4 also summarizes these items.

REFERENCES

400-1 YRP 437/93, Decommissioning Safety Analysis, September 28, 1993.

400-2 BYR 92-057, Request for Exemption From Annual and Biennial Emergency Preparedness Exercise in 1992, S. P. Schultz to M. B. Fairtile (USNRC), May 22, 1992.

400-3 NYR 92-178, Exemption From The Emergency Preparedness Rule 10CFR50.54(q) and Approval of The Defueled Emergency Plan at the Yankee Nuclear Power Station (TAC No. M83991), M. B. Fairtile (USNRC) to J. M. Grant, October 30, 1992.

400-4 EPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, US Environmental Protection Agency, October 1991.

400-5 YRC-182, Yankee Rowe Accident Atmospheric Diffusion Factors, January 5, 1982.

400-6 YRC-1024, Basis for the Radiological Status of Plant Systems and Structures, September 1993.

400-7 ELISA - Technical Description, A Computer Code for the Radiological Evaluation of Licensing and Severe Accidents at Light-Water Nuclear Power Plants, J. N. Hamawi, April 1991.

400-8 YRC-1014 Revision 1, Decommissioning Accident Source Terms.

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400-10 ESG 90/93, Flammable Gas Explosion Evaluation For Decommissioning Safety Analysis, G. A. Harper to R. A. Mellor, August 20, 1993.

400-11 NUREG/CR-2300, PRA Procedures Guide, January 1983.

400-12 NYR 92-144, Exemption From 10CFR Part 50 - Appendix E - Emergency Preparedness Training Exercises at the Yankee Nuclear Power Station (TAC No. M83415), M. B. Fairtile (USNRC) to J. M. Grant, July 24, 1992.

400-13 YRP 303/93, Impact of Wet Spent Fuel Storage on Decommissioning, P. A. Rainey to R. A. Mellor, September 27, 1993.

400-14 NUREG/CR-130, Technology, Safety, and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station, Battelle Pacific Northwest Laboratory, June 1978.

400-15 YAEC-1711, Yankee Nuclear Power Station Severe Accident Closure Submittal, December 21, 1989.

400-16 NUREG-0825, Integrated Plant Safety Assessment Systematic Evaluation Program, Yankee Nuclear Power Station, June 1983.YRC-1125, Propane Evaluation for Temporary Waste Disposal System, May 16, 1996.

400-17 EDC 96-305, “Yard Area Crane Upgrades.”

400-18 YRC-1238, DBA Curie Limits, November 17, 200

TABLE 400.1

Exclusion Area Boundary Dose per Curie Conversion Factors

	Dose Conversion Factors, rem/Ci		
	TEDE	Thyroid	Skin
Contamination: MCS and Bleed Line in VC	3.4E-02	1.4E-03	8.6E-05
Remaining Systems and Structures	2.4E-02	7.8E-04	4.9E-05
Activation: Bioshield Concrete	1.9E-03	4.1E-04	2.5E-05
Vessel Cladding	9.9E-03	2.8E-03	1.6E-04
Vessel Wall	2.1E-03	6.2E-04	3.4E-05
Bounding Dose Conversion Factors	7.36E-02	1.07E-03	5.75E-05

NOTES:

1. TEDE is total effective dose equivalent.

2. Activation results are based on the average value in the core region.

3. The Main Coolant System (MCS), Bleed Line in the Vapor Container (VC) and Reactor Vessel have been permanently removed.

4. As the radioactive source term on site decays, there are changes in the radionuclide mix available for potential release during an event. As a result of this changing mix, the dose conversion factors must be periodically updated. The Bounding Dose Conversion Factors reported in the table are the limiting values through 1/1/05.

TABLE 400.2

Summary of Materials Released in Postulated Events

Item	Radioactivity Content	Source of Radioactivity
Main Coolant System:		
Piping (1 Loop)	11 Ci	Internal Contamination
Valves (1 Loop)	1.2 Ci	Internal Contamination
Pumps (1 Pump)	3.0 Ci	Internal Contamination
Charging & Volume Control System:		
Bleed Line Piping (Total in VC)	4.6 Ci	Internal Contamination
Feed & Bleed Heat Exchanger (1 Shell)	9.5 Ci	Internal Contamination
Reactor Vessel:		
Intact Removal (Internal Contamination)	23 Ci	Internal Contamination
Segmented Removal (Cutting Debris)	120 Ci	Activated Metal
Radioactive Materials Container:		
Main Coolant System Piping Container	2.9 Ci	Internal Contamination
Reactor Vessel Segment Cask	2.3 Ci	Internal Contamination
Sea-Land Container with Combustible Material	2.9 Ci	Internal Contamination

NOTES:

1. Radioactivity content is based on January 1994.

2. The Main Coolant System, Charging & Volume Control System and Reactor Vessel have been permanently removed.

TABLE 400.3

Summary of Safety Analysis Requirements

Accident Analysis Section	Analysis Requirement
400.1.2	Special Nuclear Material used as reactor fuel will not be moved into the Reactor Vessel. Refer to Note 1.
400.1.2	The capability to isolate the Vapor Container will be retained to mitigate the consequences of a significant radioactive release. If an accident occurs, the Vapor Container will be isolated expeditiously.
403.1	Detailed planning of activities that use liquids will ensure that contaminated liquids will be processed by a liquid waste processing system. Existing or supplemental barriers will be used to ensure that inadvertent spills are contained within the liquid waste processing system.
403.1 403.3 407.7	The following will be performed if the Reactor Vessel is removed as a single component: • Internal Dross will be consolidated to stabilize contamination. • Limitations will be placed on the Reactor Vessel cask transporter operation to ensure that Spent Fuel Pit structural integrity is not adversely affected. Refer to Note 1.
403.4 406.2 407.7	Radioactive materials storage areas will be located and arranged such that multiple containers or components will not be significantly affected by a fire event causing a release of airborne radioactivity exceeding the bounding materials handling event.
403.5 407.7	The following components will be handled and transported on-site as single containers or components to reduce the consequences of a materials handling event: Reactor Vessel Casks; Main Coolant System Piping Containers, Valve Containers, and Pumps; Feed and Bleed Heat Exchanger Shells; Bleed Line Piping Containers. Refer to Note 1.
406	Explosives will not be used at YNPS without completion of a separate safety analysis. The analysis must include the effects of the use of explosives both on the Spent Fuel Pit Building and Fuel Transfer Chute structural integrity and on the potential release of airborne radioactivity.

TABLE 400.3
(Continued)

Summary of Safety Analysis Requirements

Accident Analysis Section	Analysis Requirement
406.1	The quantity of inflammable gases stored on-site will be limited as follows: • The Vapor Container and Potentially Contaminated Area Warehouse will not be used as a general storage location for inflammable gas cylinders. Only cylinders in use or required in the near term should be located inside the structures at any given time. • Acetylene cylinders used in the Vapor Container will be limited to standard size cylinders.
408	If fuel is stored in an on-site dry storage facility, the facility will be located such that it is a safe distance from decommissioning activities, precluding any significant interactions.
408.1	A separate safety analysis will be completed before movement of fuel from the Spent Fuel Pit to ensure that there are no unacceptable interactions between fuel movement and decommissioning activities.
408.2	Decommissioning activities near and around the Spent Fuel Pit Cooling System and other support systems will be controlled to prevent damage to these systems.
408.3	Activities will not be implemented that could result in the loss of Spent Fuel Pit integrity or could cause physical damage to the fuel that would reduce subcriticality margin or adversely affect the ability to cool the fuel.

NOTE:

1. The Main Coolant System, Charging & Volume Control System and Reactor Vessel have been permanently removed. This safety analysis requirement is no longer applicable to the YNPS.

TABLE 400.4

Summary of Decommissioning Planning Considerations from the Safety Analysis

Accident Analysis Section	Decommissioning Planning Considerations
401.1	The Radiation Protection Program will be applied to all activities performed on site involving radioactive materials: • Activities will be managed by qualified individuals who will implement program requirements in accordance with established procedures. • Radiation exposures and the release of radioactive materials to unrestricted areas will be maintained as far below specified limits as is reasonably achievable. • Radiation protection training will be provided to all occupationally exposed individuals.
401.1	Project management will ensure that work specifications, designs, and work packages involving radiation exposure or radioactive materials incorporate effective radiological controls.
401.2	The Occupational Safety Program will be implemented during YNPS decommissioning.
402.2	Hazardous materials handling will be controlled through the Nonradioactive Hazardous Materials Program and the Chemical Control Program.
402.2	Safe storage and use of inflammable gases and any other inflammable materials will be controlled through the Occupational Safety Program and the Fire Protection Program.
403.1 403.3 407.7	The following will be performed if the Reactor Vessel is removed as a single component: • A local HEPA filtration unit will be used as necessary to remove airborne radioactivity. • A lifting fixture/cover and nozzle covers will be installed to provide additional barriers to the release of radioactivity. Refer to Note 1.
403.2	Detailed planning will ensure that the following systems with high internal contamination will be dismantled using mechanical methods: Main Coolant System, Feed and Bleed Heat Exchanger, Bleed Line Piping. Refer to Note 1.

Accident Analysis Section	Decommissioning Planning Considerations
403.2	The following will be performed if the Reactor Vessel is segmented: • Mechanical cutting methods will be implemented to cut the Reactor Vessel. • Cutting of the Reactor Vessel beltline area will be conducted under water. A local HEPA filtration unit will be used to remove airborne radioactivity. Refer to Note 1.
403.5	Openings in components will be covered and sealed to minimize the spread of contamination after dismantlement and before on-site transportation.
404.1	Back-up power sources will be maintained to support spent fuel cooling requirements.
404.2	If service water is not available to the Vapor Container fire hose reels, an alternate source of water will be re-established in accordance with the Fire Protection Plan.
404.3	Decommissioning activities in areas ventilated by the VC Ventilation and Purge System or the Ventilation System will be suspended if ventilation capability is lost.
405	The following fire protection features will be maintained through implementation of the Fire Protection Program: • Fire detection equipment and systems • Fire barrier maintenance and control • Personnel training and qualification programs • Fire Protection Program procedures • Control of transient combustible materials and ignition sources
407.2	The impact of seismic events will be evaluated during planning of dismantlement activities. The evaluation will consider the possibility of the spread of contamination and of the impact on Spent Fuel Pit structural integrity.

NOTE:

1. The Main Coolant System, Charging & Volume Control System and Reactor Vessel have been permanently removed. This decommissioning planning consideration is no longer applicable to the YNPS.

401 EVENTS AFFECTING OCCUPATIONAL HEALTH AND SAFETY

401.1 Radiological Events

Radiological events could occur which result in increased exposure of decommissioning workers to radiation. However, the occurrences of these events are minimized or the consequences are mitigated through the implementation of the Radiation Protection Program (Section 507) and the Defueled Emergency Plan (Section 515).

The Radiation Protection Program is applied to all activities performed on site involving radioactive materials. The primary objective of the Radiation Protection Program is to protect workers and visitors to the site from radiological hazards that have the potential to develop during decommissioning. The program requires YNPS and its contractors to provide sufficient qualified staff, facilities and equipment to perform decommissioning in a radiologically safe manner.

Activities conducted during decommissioning that have the potential for exposure of personnel to either radiation or radioactive materials will be managed by qualified individuals who will implement program requirements in accordance with established procedures. Radiological hazards will be monitored and evaluated on a routine basis to maintain radiation exposures and the release of radioactive materials to unrestricted areas as far below specified limits as is reasonably achievable. Radiation protection training will be provided to all occupationally exposed individuals to ensure that they understand and accept the responsibility to follow procedures and to maintain their individual radiation dose as low as is reasonably achievable.

Project management will ensure that work specifications, designs and work packages involving potential radiation exposure or handling of radioactive materials incorporate effective radiological controls. Task planning will include consideration of the potential adverse events. The objective of this planning is to ensure that protective measures and contingency plans are developed to address the potential occurrence of these events and to minimize their impact on the workers as well as the public health and safety.

The Defueled Emergency Plan retains an on-site emergency response capability. This capability includes removal of personnel from an affected area, including site evacuation, if necessary. The plan is implemented by the control room personnel.

Implementation of these programs ensures that potential radiological events affecting occupational health and safety will be sufficiently minimized and mitigated to not warrant further consideration in this analysis.

401.2 Nonradiological Events

Decommissioning YNPS may require different work activities than were typically conducted during normal plant operations. Effective implementation of the Occupational Safety Program (Section 510) to decommissioning activities will ensure worker safety. The goal of the Occupational Safety Program is to provide a hazard-free environment for employees. The program incorporates safety into every phase of decommissioning from early design through implementation.

Implementation of the Occupational Safety Program will ensure that industrial safety events are eliminated to the maximum extent possible.

402 EVENTS AFFECTING PUBLIC HEALTH AND SAFETY

402.1 Off-Site Radiological Events

The Decommissioning Environmental Report that was prepared to support the YNPS Decommissioning Plan presented the impact of decommissioning on the public and the environs. Off-site events related to decommissioning activities are limited to those associated with the shipment of radioactive materials. Radioactive shipments will be made in accordance with all applicable requirements (e.g., NRC, Department of Transportation). The Radioactive Waste Management Program (Section 508) and the Decommissioning Quality Assurance Program (Section 514) assure compliance with these requirements.

Compliance with these requirements ensures that both the probability of occurrence and the consequences of an off-site event do not significantly affect the public health and safety.

402.2 Nonradiological Events

There are no decommissioning events that can be initiated from nonradiological sources that could significantly impact public health and safety.

Hazardous materials handling will be controlled through the Nonradioactive Hazardous Materials Program and the Chemical Control Program (Section 511). There are no chemicals stored on-site that, after release, could significantly threaten public health and safety. Inflammable gases stored on-site include combustible gases used for cutting and welding and liquid propane gas (LPG) used for operation of forklift trucks and evaporator. Safe storage and use of these gases and any other inflammable materials is controlled through the Occupational Safety Program and Fire Protection Program (Section 510 and 513).

The programs described above are implemented through procedures that control material identification, inventory, handling, storage, use and disposal, minimizing the probability of on-site nonradiological events. In addition, procedures present mitigative measures that would be implemented if an event occurred.

Implementation of these programs ensures that the probability of occurrence and consequence of on-site nonradiological events do not significantly affect public health and safety.

402.3 Radiological Events

There are no radiological events that could affect the health and safety of the public such that it results in a release of radioactive materials exceeding the EPA PAGs. Radiological events are divided into several categories: (1) decommissioning activity events, (2) loss of support system events, (3) fire events, (4) explosion events, (5) external events and (6) spent fuel storage events. Sections 403 through 408 present the dominant radiological events.

402.4 Radiological Analysis Basis

The consequences of postulated decommissioning accident scenarios on the health and safety of the public were determined by calculating the potential dose at the Exclusion Area Boundary. The

location of the Exclusion Area Boundary is presented in Figure 300-1 as a point 3100 ft from the center of the Vapor Container. The airborne release is assumed to result from release of the entire potential airborne radioactivity in a container or component. Releases to the environment are assumed to be at ground level. Activities that could result in release of radioactive liquids will be designed to contain the releases within the liquid waste processing system using existing or supplemental barriers.

An atmospheric dispersion factor of 2.84E-04 sec/m3 (Reference 400-5) was used to estimate the two hour dose at the Exclusion Area Boundary resulting from a ground level release of radioactivity. The radionuclide distributions used to evaluate postulated releases are estimated from the radiological scoping survey data. Radionuclide distributions, contamination levels, and radioactivity contents are calculated as of January 1, 1994, with the exception of the fuel handling accident which is based on May 31, 1992 (Reference 400-6). An updated calculation has been performed due to radioactive decay and changes in the radionuclide distribution over time. This update is discussed in Section 400 which also specifies the most recent calculation as reference

The ELISA computer program (Reference 400-7) was used to calculate the Exclusion Area Boundary doses relative to the radioactivity released (Reference 400-8). The following doses were calculated: total effective dose equivalent, thyroid dose, and skin dose. In each case, the total effective dose equivalent was the limiting value. Table 400.1 presents the dose to radioactivity conversion factors used in the analysis.

The off-site dose was determined based on the airborne radioactivity released in each accident scenario. Most of the calculations for the dominant scenarios used the highest dose to radioactivity conversion factors, which were based on the Main Coolant System and Bleed Line radionuclide distribution. Doses estimated for activated components were based on a combined release of loose activated base material (e.g., fine cutting debris, concrete dust) and surface contamination. Table 400.2 presents a summary of the materials that were assumed to be released.

The radiological analysis results in conservative, bounding estimates of the radiological consequences of the events considered in this analysis based on the following:

- The atmospheric dispersion factor was based on conservative meteorology. Realistic meteorology would increase dispersion, decreasing the dose by about a factor of 3 (Reference 400-7).

- The radioactivity release estimates assume that all of the radioactivity released to the environment is incorporated into a plume and is transported to the Exclusion Area Boundary. Only a fraction of the radioactivity released will form a plume and a portion of the plume will drop out prior to reaching the Exclusion Area Boundary, decreasing the dose.

- In most cases, the release fractions from the systems, structures and components were assumed to be 100% of the surface contamination and loose activated base metal. This could only result as a nonmechanistic release of radioactivity following a significant force being applied to the container or component. A significant fraction of the contamination that could be released is tightly bound to the surface. Radiological scoping analysis indicated that this fraction was between 50% to 80% (Reference 400-6). Only a fraction of the total radioactivity would be

released as a result of the energy imparted during an impact on the component or container, if the component or container is breached. More realistic release fractions of 1% to 10% are most likely justifiable. These fractions are consistent with drop, fire, and explosion scenarios in comparable evaluations (Reference 400-14).

Releases resulting from accidents postulated in the decommissioning accident analysis were evaluated against the EPA PAGs. Events producing off-site doses less than the guides were classified as not having a significant effect on the public health and safety.

403 DECOMMISSIONING ACTIVITY EVENTS

Decommissioning activities were identified on a location-by-location basis. Nine plant areas containing radiologically contaminated and activated systems, structures and components were identified. Dominant systems, structures and components were identified based on the amount of potential airborne radioactivity that could be released during decommissioning activities.

For each of the dominant systems, structures and components in each plant area, five decommissioning process steps were considered: decontamination, dismantlement, packaging, storage and materials handling.

403.1 Decontamination Events

Selected systems, structures and components will be decontaminated during decommissioning to remove radioactivity from or stabilize radioactivity on external and internal surfaces. General decontamination methods that may be applied during YNPS decommissioning are presented in Section 200. External contamination levels are significantly lower than the contamination levels on internal surfaces of systems and components. Therefore, the bounding decontamination event is based on decontamination of internal surfaces.

Internal decontamination methods typically use liquids to remove radioactivity from the surface (e.g., chemical decontamination, high pressure water washing). Detailed planning of decommissioning activities that use liquids will ensure that contaminated liquids will be processed by the liquid waste processing system. Additionally, existing or supplemental barriers will be used to ensure that inadvertent spills from these activities will be contained within the liquid waste processing system. These precautions prevent an unmonitored release of radioactive liquids to the environment.

If the Reactor Vessel is removed as a single component (Section 201.2), the internal dross will be consolidated while the vessel is drained. A local HEPA filtration unit will be used as necessary to remove airborne radioactivity generated during the vessel removal preparation. Additional, routine air sampling performed as part of the Radiation Protection Program would detect increased radioactivity in the Vapor Container atmosphere and the inadvertent release of radioactivity would be identified and stopped.

A bounding analysis was completed for the purposes of this accident analysis to estimate the consequences of generating airborne radioactivity during decontamination activities. The dominant system, structure or component that could have caused the highest off-site dose as a result of a release of airborne radioactivity during decontamination was the Reactor Vessel. The bounding analysis conservatively assumed that all of the radioactivity in the contamination layer on the internal surface of the Reactor Vessel was nonmechanistically released to the Vapor Container atmosphere.

Based on the evaluation of a gas bottle explosion event in the Vapor Container presented in Section 406, the amount of radioactivity that would be transported from the Vapor Container to the environment following the event was about 10%. The motive forces associated with the release of

contamination during a decontamination event are significantly less than those during an explosion. However, an instantaneous release of 10% of the potential airborne radioactivity from the component to the environment was used as a bounding value for the purposes of the calculation.

The estimated radioactivity content of the Reactor Vessel internal surface contamination was about 8 Ci. A release of this material to the environment would result in an off-site dose at the Exclusion Area Boundary of about 0.28 rem. This is significantly less than the EPA PAGs. In addition, radiological scoping analyses indicated that between 50% and 80% of the internal surface contamination is tightly bound. Incorporation of this effect would reduce the dose to less than 0.14 rem.

The radiological consequences of the bounding decontamination event result in a dose at the Exclusion Area Boundary significantly less than the EPA PAGs. Therefore, the public health and safety are not significantly affected by the potential decontamination events that could occur during decommissioning.

403.2 Dismantlement Events

Systems, structures and components will be dismantled during decommissioning to remove radioactive materials from the site. General dismantlement methods that may be applied during YNPS decommissioning are presented in Section 200. Detailed planning will ensure that systems with high internal contamination are dismantled using mechanical methods (e.g., split frame machining). This restriction limits the amount of airborne contamination generated during dismantlement activities. Thermal dismantlement methods (e.g., plasma arc, oxy-fuel) may be used on systems with lower radioactivity levels.

If the Reactor Vessel is segmented, a combination of cutting methods will be implemented: metal disintegration machining and milling (Section 201.2). Most of the Reactor Vessel cutting will be conducted under water, which minimizes the generation of airborne radioactivity. In addition, local HEPA filtration will be used to remove any gases and airborne radioactivity generated during cutting operations.

The portion of the Reactor Vessel above the vessel support lugs will be segmented above water. Prior to cutting, the internal surface will be decontaminated to remove loose contamination. The radioactivity from neutron activation of the vessel wall in this region is significantly lower compared to the core region. A contamination envelope with HEPA filtration will be used to preclude release of airborne radioactivity to the Vapor Container atmosphere.

A bounding analysis was completed for the purposes of this accident analysis to estimate the consequences of generating airborne radioactivity during dismantlement activities. The dominant system, structure or component that could have caused the highest off-site dose as a result of a release of airborne radioactivity during dismantlement was one Main Coolant System Loop. The bounding analysis conservatively assumed that all of the radioactivity on the internal surface of one Main Coolant System Loop was nonmechanistically released to the Vapor Container atmosphere. This event is bounding because underwater cutting of the highest radioactivity regions of the Reactor Vessel reduces the potential airborne contamination generation from that source.

Based on the evaluation of a gas bottle explosion event in the Vapor Container presented in Section 406, the amount of radioactivity that would be transported from the Vapor Container to the environment following the event was about 10%. The motive forces associated with the release of contamination during a dismantlement event are significantly less than those during an explosion. However, an instantaneous release of 10% of the potential airborne radioactivity from the Vapor Container to the environment was used as a bounding value for the purposes of the calculation.

The estimated radioactivity content of a Main Coolant System Loop internal surface contamination was about 15 Ci. A release of 10% of this material to the environment would result in an off-site dose at the Exclusion Area Boundary of about 0.051 rem. This is significantly less than the EPA PAGs. In addition, radiological scoping analyses indicated that between 50% and 80% of the internal surface contamination is tightly bound. Incorporation of this effect would reduce the dose to less than 0.026 rem. A nonmechanistic release of the cuttings from segmentation of the upper portion of the Reactor Vessel segmentation would have resulted in an off-site dose of less than 0.013 rem.

The radiological consequences of the bounding dismantlement event result in a dose at the Exclusion Area Boundary significantly less than the EPA PAGs. Therefore, the public health and safety are not significantly affected by the potential dismantlement events that could occur during decommissioning.

403.3 Packaging Events

Radioactive materials are packaged prior to shipment from YNPS to either a disposal facility or an off-site processing facility. Intermediate packaging may also be used prior to transporting radioactive materials from their removal area to a final packaging area. The materials handling event presented in Section 403.5 nonmechanistically assumes that the potential airborne radioactivity in a bounding container or component is released directly to the environment. This event also bounds any packaging event.

If the Reactor Vessel is removed as a single component, the vessel will be packaged in a shipping cask located directly under the Vapor Container Equipment Hatch. Internal dross will be consolidated prior to vessel lift. A lifting fixture/cover and nozzle covers will be installed to provide additional barriers to the release of radioactivity. These considerations preclude any significant release of radioactivity during packaging of the Reactor Vessel.

The radiological consequences of a packaging event are bounded by the consequences of a materials handling event. Therefore, the public health and safety are not significantly affected by the potential packaging events that could occur during decommissioning.

403.4 Storage Events

Containers and components will be stored on-site prior to shipment to either a disposal facility or an off-site processing facility. Intermediate storage locations may also be used before the radioactive materials are moved to a packaging area. Several evaluations are presented in the accident analysis regarding the storage of radioactive materials:

- Fire Events (Section 405)
- Explosion Events (Section 406)
- External Events (Section 407)

Each of these sections presents restrictions to ensure that adequate separation by barrier, distance or radioactivity content is employed to preclude an event causing a release that exceeds the bounding materials handling event presented in Section 403.5. The materials handling event presented in Section 403.5 nonmechanistically assumes the breach of a single container or component, which releases the total potential airborne radioactivity directly to the environment. Storage areas will be located and arranged such that multiple containers or components could not be affected by a single event causing a release of airborne radioactivity that exceeds the bounding materials handling event.

The radiological consequences of a storage event are bounded by the consequences of a materials handling event. Therefore, the public health and safety are not significantly affected by the potential storage events that could occur during decommissioning.

403.5 Materials Handling Events

Materials handling events encompass those events that could potentially occur during movement of radioactive materials from their removal location to a staging location outside of the structure containing the materials. Subsequent handling of these materials is considered by the on-site transportation external event presented in Section 407.7.

After removal, all openings in components will be covered to minimize the spread of contamination. Components will then either be placed in containers for on-site transportation or be transported individually. The following components containing high potential airborne radioactivity were handled as single containers or components to reduce the consequences of a materials handling event: Reactor Vessel Casks; Main Coolant System Piping Containers, Valve Containers, and Pumps; Feed and Bleed Heat Exchanger Shells; Vapor Container Bleed Line Piping Containers.

The non-mechanistic release of the contents of the feed and bleed heat exchanger described below (for historical purposes), establishes a bounding dose of 320 mrem TEDE for a materials handling event. The feed and bleed heat exchanger has now been removed from the site. New dose conversion factors have been determined based on limiting radionuclide mixes and are presented in Table 400.1. These bounding dose conversion factors may be used to determine the concentration of radioactivity in a single container that would have the potential to produce 320 mrem at the Exclusion Area Boundary from an accidental release.

A bounding analysis was completed for the purposes of this accident analysis to estimate the consequences of generating airborne radioactivity resulting from a materials handling event. The dominant system, structure or component that could have caused the highest off-site dose as a result of a release of airborne radioactivity during handling was one of the four Feed and Bleed Heat Exchanger shells. The bounding analysis conservatively assumed that all of the radioactivity on the internal surfaces of a Feed and Bleed Heat Exchanger shell was nonmechanistically released to the environment. If the Reactor Vessel is removed as a single component, internal dross will be consolidated to stabilize the contamination. This would significantly reduce the amount of radioactivity that could be released during handling.

The estimated radioactivity content of one Feed and Bleed Heat Exchanger shell internal surface contamination was about 9.5 Ci. A release of this material to the environment would result in an off-site dose at the Exclusion Area Boundary of about 0.320 rem. This is significantly less than the EPA PAGs. In addition, radiological scoping analyses indicated that between 50% and 80% of the internal surface contamination was tightly bound. Incorporation of this effect would have reduced the dose to less than 0.160 rem. The nonmechanistic release assumed that the heat exchanger shell failed catastrophically, releasing radioactivity from all surfaces to the environment. Realistically, the shell is structurally stable and total failure is highly unlikely, further reducing the release of radioactivity.

The following are the potential off-site doses at the Exclusion Area Boundary for the nonmechanistic release of the radioactivity from high radioactivity containers presented above:

Vapor Container Bleed Line Piping	0.160 rem
Main Coolant Pump (1)	0.100 rem
Main Coolant Pipe Container	0.100 rem
Reactor Vessel Segment Cask	0.078 rem
Main Coolant Valve Container	0.041 rem

The radiological consequences of the bounding materials handling event result in a dose at the Exclusion Area Boundary significantly less than the EPA PAGs. Therefore, the public health and safety are not significantly affected by the potential materials handling events that could occur during decommissioning.

404 LOSS OF SUPPORT SYSTEM EVENTS

The electric power, cooling water and compressed air systems provide support to both spent fuel storage and decommissioning activities. Loss of these systems could potentially affect many systems and plant areas simultaneously. However, none of these support systems are safety related.

404.1 Loss of Off-Site Power

Off-site power is used to energize components of the Spent Fuel Pit Cooling System and Fuel Transfer Enclosure and to energize tools, cranes, lighting and air filtering equipment used during decommissioning activities. The following results from a loss of off-site power:

- Spent fuel cooling capability is lost. The consequences of a loss of spent fuel cooling capability are presented in Section 408.

- Decommissioning tools, lighting and air filtering equipment are de-energized. All decommissioning activities will be terminated.

- Cranes are de-energized and lifting operations are terminated. Cranes fail in a safe condition when they are de-energized.

Back-up power sources will be maintained to support spent fuel cooling requirements, however, back-up power is not needed to support decommissioning activities.

Loss of off-site power will not result in the failure of containment systems designed to preclude the spread of contamination (e.g., local contamination control envelopes, HEPA filters). Although the HEPA filter fans will stop, the filter will remain intact and the contamination control envelope will not be breached, preventing unfiltered releases. A breach of the contamination envelope is an independent event and is not considered in this evaluation. Any significant breach of the contamination envelope would be detected and closed independent of a loss of off-site power.

A loss of off-site power does not result directly in a release of radioactive material to the environment during decommissioning activities. Section 408 demonstrates that the public health and safety are not adversely affected from a loss of spent fuel cooling capability. Therefore, public health and safety are not adversely affected by a loss of off-site power event.

404.2 Loss of Cooling Water

Cooling water is supplied by the Auxiliary Service Water System to a heat exchanger which cools the Spent Fuel Pit Cooling System. As a result of loss of cooling water, spent fuel cooling capability is lost. The consequences of a loss of spent fuel cooling capability are presented in Section 408.

A loss of cooling water does not result directly in a release of radioactive material to the environment during decommissioning activities. Section 408 demonstrates that the public health and safety would not be adversely affected from a loss of spent fuel cooling capability. Therefore, public health and safety are not adversely affected by a loss of cooling water event.

404.3 Loss of Compressed Air

Compressed air is supplied by various air compressors to operate fuel handling equipment and power pneumatic tools. The following occurs following a loss of compressed air:

- Fuel handling equipment fails in a safe condition. Compressed air is required to unlatch a load from the fuel handling tool.

- Decommissioning pneumatic tools shut down. This stops any potential releases from these activities.

The radiological consequences of a loss of compressed air event are small and are bounded by the materials handling event presented in Section 403.5. Therefore, public health and safety are not adversely affected by a loss of compressed air event.

405 FIRE EVENTS

A fire event could affect several plant systems, structures and components simultaneously. Combustible materials can be ignited by either external ignition sources (e.g., oxyacetylene torches) or internal ignition sources (e.g., spontaneous combustion). Adequate levels of the following fire protection features will be maintained through implementation of the Fire Protection Program (Section 513) minimizing the potential of occurrence of a fire:

- Fire detection equipment and systems.
- Personnel training and qualification programs.
- Fire Protection Program procedures.
- Control of transient combustible materials and ignition sources.

In addition, if a fire occurs, the following fire protection features will be employed to limit the consequences to those of the decommissioning materials handling event presented in Section 403.5:

- Maintain sufficient fire detection, response and suppression capability.
- Separate containers, as necessary, by distance, barrier or radioactivity content.

Higher radioactivity containers are unlikely to fail and cause a significant release due to a fire as these containers are designed for greater levels of integrity. Spontaneous combustion inside a container is highly unlikely as most containers are filled with noncombustible materials. Sea-land containers may be used to ship combustible radioactive materials. The estimated radioactivity level of a sea-land container filled with combustible radioactive material is about 2.9 Ci. Release of all of the radioactivity as a result of a fire would result in an off-site dose of 0.100 rem at the Exclusion Area Boundary. The assumption of a total release of radioactivity is very conservative. Reference 400-14 presents a release fraction of 0.00015 for a similar event. Incorporation of this assumption would significantly reduce the release.

Implementation of the Fire Protection Program minimizes the probability of occurrence of a fire. Implementation of the restrictions presented above limits radiological consequences at the Exclusion Area Boundary to a value significantly less than the EPA PAGs. Therefore, public health and safety are not adversely affected by a fire event.

406 EXPLOSION EVENTS

An explosion event could affect several plant systems, structures and components simultaneously. Explosions are possible from the following sources:

- Ion Exchange Resin Offgases
- Explosives
- Inflammable Gas Storage Bottles and Tanks

Processing and limitation of ion exchange resin offgases will continue to be controlled by procedures that have successfully precluded this type of accident throughout the greater than 30-year operation of YNPS. Explosives will not be used at YNPS without completion of a separate safety analysis. The analysis must include the effects of the use of explosives both on the Spent Fuel Pit Building and Fuel Transfer Chute structural integrity and on the potential release of airborne radioactivity.

Inflammable gases (e.g., acetylene, LPG) may be used during decommissioning for thermal cutting or to power material handling and processing equipment. Inflammable gas cylinders will be used, located and stored in quantities such that the possibility of explosion is minimized. Two 1,000 gallon propane tanks, located approximately 400 feet from the Spent Fuel Pit Building, supply the Temporary Waste Water Processing Island System’s gas-fired boiler. The quantities of the inflammable gas cylinders and the location of the propane tanks will be such that the radiological consequences of an explosion are bounded by the events described below.

406.1 Explosion Events: Vapor Container

Inflammable gases may be used in the Vapor Container during thermal cutting activities. The Decommissioning Plan recommends mechanical cutting of significantly contaminated systems, structures and components to minimize the generation of airborne radioactivity. However, thermal cutting methods utilizing inflammable gases could be used for dismantlement activities on components with lower contamination levels.

An engineering evaluation was completed to determine the physical consequences of an explosion in the Vapor Container (Reference 400-10). If the Vapor Container is closed, the peak pressure resulting from an explosion of a standard cylinder of acetylene is less than 5% of the Vapor Container design pressure. If the Vapor Container is not isolated (e.g., Equipment Hatch open), about 10% of the Vapor Container air mass could be released through existing openings before the pressure inside the Vapor Container equalizes with the outside atmospheric pressure. It is assumed that radiological contamination controls are re-established after the explosion by closing Vapor Container openings to terminate the release to the environment.

The explosion evaluation assumes that all of the energy from the gas explosion is converted into a pressure increase. This is a highly conservative assumption; most of the energy would be lost to other effects (e.g., heat generation, mechanical deformation). Less than 5% of the energy most likely would be converted into the pressure increase.

If an explosion nonmechanistically releases all of the internal surface contamination contained in systems inside the Vapor Container to the Vapor Container atmosphere, the release to the

environment would not exceed the EPA PAGs at the Exclusion Area Boundary. The radioactivity content of the internal surface contamination in systems inside the Vapor Container is less than 130 Ci. Release of 10% of this material to the environment would result in a dose at the Exclusion Area Boundary of about 0.440 rem.

It is highly unlikely that all systems either will be opened or will be in the Vapor Container such that an explosion could remove the internal radioactivity. In addition, a significant fraction of the internal surface contamination is tightly bound. Radiological scoping analyses indicated that this fraction varied between 50% and 80%. The combination of these effects would reduce the release by 70% to 90%, resulting in an off-site dose of less than 0.100 rem.

In order to minimize the overall risk of an inflammable gas-air explosion in the Vapor Container, the Vapor Container should not be used as a general storage location for inflammable gas cylinders. Only cylinders in use or required in the near term should be located inside the structure at any given time. Additionally, thermal cutting methods using inflammable gases were not used to dismantle the following components with high contamination levels: Reactor Vessel, Main Coolant System, Feed and Bleed Heat Exchanger, Vapor Container Bleed Line Piping.

The consequences of an explosion in the Vapor Container are significantly less than the EPA PAGs. Therefore, there is no significant impact on pubic health and safety.

406.2 Explosion Events: Potentially Contaminated Area Warehouse

Inflammable gases may be used in the Potentially Contaminated Area Warehouse to power forklift trucks and to support thermal cutting activities. In the unlikely event that a single inflammable gas cylinder explodes in the warehouse, the pressure would be in excess of the capacity of the warehouse if the structure is sealed. Doors, portions of the roof or portions of the walls would likely fail in order to relieve excess pressure. Explosion released overpressure would not cause the containers to explode since the explosion is external to the containers. Some damage of the containers would be likely, but a release of significant portions of the contents of the containers as airborne radioactivity is not likely. In addition, the segregation and separation practices implemented for fire protection purposes would also limit the effects on storage containers.

In order to minimize the overall risk of an inflammable gas-air explosion in the Potentially Contaminated Area Warehouse, the warehouse should not be used as a general storage location for inflammable gas cylinders. Only cylinders in use or required in the near term should be located inside the structure at any given time.

The consequences of an explosion in the Potentially Contaminated Area Warehouse are bounded by those of the decommissioning materials handling events presented in Section 403.5. Therefore, there is no significant impact on pubic health and safety.

407 EXTERNAL EVENTS

A systematic assessment of external events was made to evaluate the effects of natural and manmade events on decommissioning activities. The hazards associated with these events are assumed to be consistent with those that could have occurred while YNPS was in operation. Seven external events were identified as having potential applicability to the YNPS decommissioning based on a review of the natural and manmade external events presented in NUREG/CR-2300 (Reference 400-11).

407.1 Aircraft Impact

YAEC evaluated the significance of a potential aircraft hazard on YNPS in response to Systematic Evaluation Program Topic III-4 (Reference 400-16). The analysis concluded that the annual probability of an aircraft impact was very low. Further consideration of the interaction between an aircraft impact and decommissioning is not warranted.

407.2 Earthquake

A seismic event during decommissioning could initiate a materials handling event similar to those described in Section 403.5. The analysis in Section 403.5 concludes that the bounding material handling event results in an off-site dose that is significantly less than the EPA PAGs. In addition, detailed planning of dismantlement activities will consider the impact of seismic events on components that are affected by removal activities. These components will be evaluated and physically supported, as appropriate, to limit the off-site dose resulting from a release of radioactivity to less than the EPA PAGs.

The Spent Fuel Pit and Fuel Transfer Chute were analyzed for seismic considerations and found acceptable as part of the Systematic Evaluation Program. The Yard Area Crane and crane support structure were analyzed for seismic considerations, including a seismic event with a loaded fuel cask suspended, and found acceptable as part of the Yard Area Crane Upgrade (Reference 400-18). All structures whose failure during a seismic event that could significantly affect either the Spent Fuel Pit structural integrity or the spent fuel integrity are seismically qualified. The impact of seismic events will be evaluated during planning of dismantlement activities in proximity of the Spent Fuel Pit. The purpose of the evaluation is to ensure that partial dismantling of equipment and structures does not result in a configuration that could fail during a seismic event, subsequently collapsing onto or into the Spent Fuel Pit or Fuel Transfer Chute.

The potential for radiological consequences from a seismic event resulting in a dose at the Exclusion Area Boundary greater than the EPA PAGs is extremely low. In the unlikely event that a materials handling event is initiated, the consequences would be significantly less than the EPA PAGs. Therefore, the public health and safety are not significantly affected by a potential seismic event during decommissioning.

407.3 External Flooding

A flooding event at YNPS typically would be preceded by a sufficient warning period to prepare the site for the event by securing decommissioning activities. Most of the potentially removable radioactivity at YNPS is located in the Vapor Container, well above the potential flood height. Most

of the balance of contaminated systems, structures and components would either be packaged for shipment or secured inside buildings. Containers that hold high radioactivity materials are designed for greater levels of structural integrity, providing additional protection. In the unlikely event that a lower radioactivity container or component is exposed to flood waters and radioactive material is dispersed, the flooding dilution effect results in a radiological consequence significantly less than an airborne release of a similar amount of radioactivity.

Flooding could initiate a loss of off-site power event. The analysis in Section 404.1 concludes that public health and safety are not adversely affected from a loss of off-site power event.

Flooding events at YNPS do not result in a significant radiological release, therefore, public health and safety are not adversely affected.

407.4 Tornadoes and Extreme Winds

The annual strike probability of a tornado that could cause a significant release of radioactivity from a container or component is very low. In addition, most components and containers that would be vulnerable to a tornado will be packaged awaiting shipment. The integrity of these containers would limit the probability and consequences of a significant release of radioactivity. Further consideration of the interaction between a tornado and decommissioning is not warranted.

An extreme winds event (i.e., hurricanes above 74 mph, thunderstorms above 100 mph) at YNPS typically would be preceded by a sufficient warning period to prepare the site for the event by securing decommissioning activities. Most of the potentially airborne radioactivity at YNPS is located in the Vapor Container, which protects the components from the effects of extreme winds and associated missiles. Most of the balance of contaminated systems, structures and components would either be packaged for shipment or secured inside buildings with system integrity, providing protection from the extreme winds and associated missiles.

Containers that hold higher radioactivity materials are designed for greater levels of structural integrity, providing additional protection. In the unlikely event that a lower radioactivity container or component is unprotected and is exposed to extreme winds and radioactive material is dispersed, the combination of low radioactivity content and significant dispersion by the wind would result in an off-site dose that is bounded by the limiting release of the materials handling event presented in Section 403.5.

Tornadoes or extreme winds could initiate a loss of off-site power event. The analysis in Section 404.1 concludes that public health and safety are not adversely affected from a loss of off-site power event.

Extreme tornadoes and winds events at YNPS will not result in a significant radiological release, therefore, public health and safety are not adversely affected.

407.5 Forest Fire

A forest fire event at YNPS typically would be preceded by a sufficient warning period to prepare the site for the event by securing decommissioning activities and closing Vapor Container openings,

if necessary. In addition, YNPS is protected from the effects of a forest fire by a buffer zone surrounding the facility. The probability of a forest fire of sufficient intensity to bridge the zone is very low (Reference 400-15).

In the unlikely event that a forest fire causes an on-site fire, the fire protection features described in Section 405 would be sufficient to mitigate the consequences of the fire. A forest fire could also initiate a loss of off-site power event. The analysis in Section 404.1 concludes that public health and safety are not adversely affected from a loss of off-site power event.

Forest fire events at YNPS will not result in a significant radiological release, therefore, public health and safety are not adversely affected.

407.6 Lightning Event

The lightning strike annual probability for a decommissioning activity or single exposed container or component is very low. In addition, YNPS structures provide protection against lightning. Although the effects of lightning generally are localized, a lightning strike could initiate a loss of off-site power event or a fire. The analyses in Sections 404.1 and 405 conclude that public health and safety are not adversely affected from a loss of off-site power event or fire event. Further consideration of the interaction between decommissioning and a lightning event are not warranted.

407.7 On-Site Transportation Accidents

On-site transportation accident events consist of those events occurring after removal of a container or components from a structure, but before final packaging for shipment. These events could occur during transportation of a container or component to a packaging area.

Detailed planning will ensure that only one container or component with high potential airborne radioactivity is transported simultaneously by the same vehicle until they are prepared for final shipment. This constraint ensures that the off-site dose resulting from an on-site transportation event is bounded by the materials handling event presented in Section 403.5. Additionally, detailed planning will limit the maximum number of containers transported on-site to ensure that they are also bounded by this analysis.

Containers and components stored in materials storage areas awaiting final packaging will be located to preclude impact from a runaway vehicle. The location selection will consider separation by distance, barrier or radioactivity to ensure that the off-site dose resulting from an impact is bounded by the dose resulting from the materials handling event presented in Section 403.5.

If the Reactor Vessel is removed as a single component, it will be transported in a cask in close proximity of the Spent Fuel Pit. Impact of the Reactor Vessel cask or the transporter with the Spent Fuel Pit could challenge the structural integrity of the Spent Fuel Pit. The following events associated with movement of the Reactor Vessel Cask were considered:

- Transporter Failure From Flat Tires - Failure of the tires on one side of the transporter could affect load stability. However, a flat tire would be isolated to one tire on an axle that has several other tires. The loss of one tire would not cause the Reactor Vessel cask to tilt such that the load

becomes unstable. Common mode failure of all tires on one side of the transporter is not credible.

- Transporter Failure From a Loss of Hydraulic Compensation - Two parallel hydraulic compensation trains extend the full length of the transporter. A failure of one train would cause the transporter flatbed to tilt. Limits will be placed on the maximum elevation of the flatbed above its minimum elevation (zero pressure configuration) during travel near the Spent Fuel Pit to ensure that the load remains stable.

- Transporter Collision With Spent Fuel Pit Wall - A collision of the transporter into the Spent Fuel Pit wall could adversely affect the structural integrity of the wall. Limits will be placed on the speed of the transporter to limit the kinetic energy of impact to a level ensuring that the wall is not damaged structurally.

The radiological consequences of an on-site transportation event are bounded by a decommissioning materials handling event. The physical consequences of an on-site transportation event does not adversely affect the Spent Fuel Pit structural integrity. Therefore, public health and safety are not adversely affected by an on-site transportation event.

408 SPENT FUEL STORAGE EVENTS

This section evaluates potential accident scenarios that could impact spent fuel storage during decommissioning. The events below correspond to storage of spent fuel in the Spent Fuel Pit. If fuel is stored in an on-site dry storage facility, the facility will be located such that it is a safe distance from decommissioning activities, precluding any significant interactions.

408.1 Fuel Handling Event

YAEC re-evaluated the Spent Fuel Pit fuel handling event during preparation for the transition to a possession only license (Reference 400-2). The analysis demonstrated that the release resulting from a fuel handling event would be significantly less than the EPA PAGs. This analysis bounds any fuel handling event that could occur during decommissioning. The NRC performed an independent review of the analysis, confirming the results (Reference 400-12). The results of the analysis are summarized below.

The fuel handling accident in the Spent Fuel Pit assumes that a fuel assembly is dropped, strikes the floor, rotates towards the horizontal, and strikes a sharp object in the most vulnerable area. The analysis applied the available kinetic energy as a linear point force to the fuel pins and assumed that the pins would fail when 1% of the yield stress is exceeded. No credit was taken for the fuel pellets within the fuel pins (i.e., the pins are assumed to be hollow). For this case, approximately 6 rows (100 pins) were estimated to fail. However, for the radiological analysis, it was conservatively assumed that all of the fuel pins in one assembly failed.

The fuel handling accident TEDE dose at the Exclusion Area Boundary is less than 0.001 rem. The radionuclide inventory used in the estimate was based on the worst case inventory on May 31, 1992. The worst case radionuclide inventory will continue to decrease due to radioactive decay.

Movement of fuel from the Spent Fuel Pit to either an on-site or off-site fuel storage facility is not bounded by this evaluation. A separate evaluation is needed to ensure that there are no unacceptable consequences during movement from the Spent Fuel Pit. This evaluation must also include any interactions between fuel movement and decommissioning activities. The evaluation will be completed before movement of fuel out of the Spent Fuel Pit.

The radiological consequences of the fuel handling accident result in a dose at the Exclusion Area Boundary significantly less than the EPA PAGs. Therefore, the public health and safety are not significantly affected by a fuel handling accident.

408.2 Loss of Spent Fuel Cooling Capability

Although spent fuel cooling capability must be maintained, the consequences of a loss of that capability are not severe. As of January 1994, greater than four weeks must elapse without re-establishing cooling or adding make-up water before the water remaining in the Spent Fuel Pit is insufficient to provide adequate shielding. A make-up water flow rate of about 1 gpm is needed to replace water lost through evaporation.

Several diverse sources of make-up are available including demineralized water, fire water, auxiliary service water or Sherman Reservoir water. Water may be injected to the Spent Fuel Pit through installed or portable pumps as well as gravity feed. Adequate time is available to maintain Spent Fuel Pit inventory.

Although scenarios that result in a loss of spent fuel cooling capability can be mitigated without any significant consequence, cooling capability will be protected during decommissioning. Decommissioning activities near and around the Spent Fuel Pit Cooling System and other support systems will be controlled to prevent damage to these systems. This may be accomplished by physically protecting the systems or by establishing safe load paths and protective zones around the systems.

The time available to respond to a loss of spent fuel cooling capability is sufficient to ensure that the event is terminated precluding any impact on the public health and safety. Implementation of the restrictions presented above will reduce the probability of occurrence of this event.

408.3 Interactions Between Spent Fuel and Decommissioning

YAEC has evaluated the option to store fuel in the Spent Fuel Pit during the decommissioning dismantlement phase (Reference 400-13). The evaluation identified safety considerations and limitations on decommissioning activities associated with operating the Spent Fuel Pit concurrent with dismantlement activities.

Potential dismantlement activities are limited when spent fuel is stored in the Spent Fuel Pit. Activities cannot be implemented that could result in the loss of Spent Fuel Pit integrity or could cause physical damage to the fuel that would reduce subcriticality margin or adversely affect the ability to cool the fuel.

Adverse spent fuel and decommissioning interactions can be precluded by incorporating the following restrictions:

- Delay dismantlement of the Ion Exchange Pit until after the fuel is permanently removed from the Spent Fuel Pit.

- The Yard Area Crane has been upgraded to a single-failure-proof crane meeting the requirements of NUREG-0554 with the implementation of EDC 96-305, "Yard Area Crane Upgrades." Technical Specification 3.2 currently limits cask usage in and over the Spent Fuel Pit to a shipping or transfer cask weighing no more than 80 tons.

- Isolate the Fuel Transfer Chute from the Spent Fuel Pit by (a) resupporting the Fuel Transfer Chute/Spent Fuel Pit penetration assembly to the Spent Fuel Pit using the latch mechanism, (b) filling the annular space between the Fuel Transfer Chute pipe and the Spent Fuel Pit penetration pipe with grout, (c) removing one section of Fuel Transfer Chute pipe uphill of the Lower Lock Valve (LLV), (d) installing a blind flange cap on the LLV, (e) erecting permanent form work and placing a concrete barrier in the LLV pit and (f) installing metal plates above and below the LLV pit to preclude personnel access to this area. This has been completed with the implementation of EDC 95-303, "Fuel Transfer Chute Isolation."

- Ensure that detailed work planning excludes activities that could result in a drop of a heavy load onto or into the Spent Fuel Pit or Fuel Transfer Chute. Ensure that partial dismantling of equipment and structures does not result in a configuration that could result in failure during an external event (e.g., seismic event) and subsequent collapse onto or into the Spent Fuel Pit or Fuel Transfer Chute. Alternatively, consider modifications to protect the Spent Fuel Pit and the Fuel Transfer Chute from heavy load drops. Technical Specification 3.2 limits movement of loads over the Spent Fuel Pit to those less than 900 pounds, other than for the specific exceptions that have been identified.

- Ensure that demolition explosives that could affect the Spent Fuel Pit structural integrity are not permitted for use until either fuel is permanently removed from the Spent Fuel Pit or an analysis of the impact of explosives on the Spent Fuel Pit structure is completed.

Incorporation of the restrictions presented above is sufficient to preclude significant events resulting from interactions between fuel storage and decommissioning activities. There is no significant effect on public health and safety.

500 ADMINISTRATION OF THE DECOMMISSIONING PLAN

500.1 Regulatory Basis for Administration of the Decommissioning Plan

Draft Regulatory Guide DG-1071, “Standard Format and Content for Post-Shutdown Decommissioning Activities Report)” (Reference 500-6), encourages licensees with an approved Decommissioning Plan to “extract the pertinent detail from the Decommissioning Plan and submit a PSDAR update in the format and content specified by the regulatory guide.” As a result, information from the approved YNPS Decommissioning Plan that was incorporated into the FSAR has since been extracted, updated and relocated to the PSDAR within the FSAR.

REFERENCES

500-1 Letter, YAEC to USNRC, dated December 20, 1993.

500-2 Draft Regulatory Guide DG-1005, “Standard Format and Content For Decommissioning Plans for Nuclear Reactors.”

500-3 Letter, USNRC to YAEC, dated February 14, 1995.

500-4 Letter, YAEC to USNRC, dated August 22, 1995.

500-5 NUREG-0586, Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities, August 1988.

500-6 Draft Regulatory Guide DG-1071, “Standard Format and Content for Post-Shutdown Decommissioning Activities Report.”

501 DECOMMISSIONING COST ESTIMATE AND FUNDING PLAN

501.1 Decommissioning Cost Estimate

This information has been extracted, updated and relocated to the PSDAR within the FSAR.

501.2 Decommissioning Funding

This information has been extracted, updated and relocated to the PSDAR within the FSAR.

502 DECOMMISSIONING ORGANIZATION AND RESPONSIBILITIES

502.1 YAEC Commitment

Yankee Atomic Electric Company (YAEC) is committed fully to compliance with the existing license and applicable regulatory requirements during all phases of YNPS decommissioning. YAEC's commitment to the safe decommissioning of the facility will be accomplished with diligence and quality. Corporate principles, policies, and goals will be followed to ensure performance excellence, management competence, and high standards in every facet of the decommissioning.

502.1.1 Goals

The primary goals of the YNPS decommissioning are 1) to safely remove the nuclear facilities from service and to reduce residual radioactivity to a level that permits release of the property for unrestricted use and termination of the license and 2) to safely store spent nuclear fuel until it can be removed from the site. While achieving these primary goals, YAEC will conduct all decommissioning operations consistent with applicable regulations and a focus on the following considerations:

- Maintain radiation exposure to the public and on-site personnel as low as is reasonably achievable.
- Ensure occupational safety for all on-site personnel.
- Minimize environmental impact.
- Minimize radioactive waste generation.
- Ensure prudent expenditure of decommissioning funds.

502.1.2 Organizational Strategy

The organizational strategy must ensure that adequate numbers of experienced and knowledgeable personnel are available to perform the technical and administrative tasks required to decommission YNPS. The YNPS decommissioning organization is currently staffed with experienced individuals who have worked at or are very knowledgeable of the plant. YAEC believes that retaining personnel with intimate knowledge of YNPS is important to the success of decommissioning.

YAEC is responsible for YNPS decommissioning. In this position YAEC has direct control and oversight over all decommissioning activities. This role is similar to that taken by YAEC during the 31 year operation of YNPS. In that role YAEC provided operational, technical, licensing, and project management support of YNPS. YAEC will contract services to supplement its capabilities as necessary.

Organizational impacts associated with the transition from plant closure to dismantlement were evaluated periodically to assure that adequate and appropriate staffing levels and capabilities were

preserved. YAEC will maintain staffing, to the extent possible, to optimize resource utilization and to capitalize on positive project momentum. Both of these are characteristics of successful project teams.

Technical Specification 6.3.1 requires that YNPS management personnel meet or exceed the minimum qualifications for education, training, and experience outlined in ANSI N18.1-1971, for comparable positions.

502.2 YAEC Organization and Functions

The YAEC organizational structure that will be implemented during decommissioning is presented in Section 1 of the Yankee Decommissioning Quality Assurance Program (YDQAP) Manual. Decommissioning functions are described in site administrative procedures.

503 TRAINING PROGRAM

YAEC will maintain a training program commensurate with the needs of the various phases of decommissioning and the Training Rule 10CFR50.120. The training program, in conjunction with other administrative programs and controls, will ensure that qualified individuals are available to operate and maintain the facility in a safe manner. The training programs will be based on a systematic analysis of job performance requirements. The analysis will ensure that personnel will have qualifications commensurate with the performance requirements of their jobs.

503.1 General Employee Training

General Employee Training will be provided to all personnel who have unescorted access to the YNPS site. Initial and annual requalification training will include the following topics commensurate with the stages of decommissioning which typically will consist of information on the following topics:

- Plant Organization and Administration
- Plant Description
- Occupational Safety
- Quality Assurance
- Fire Protection
- Emergency Response
- Radiation Protection
- Security

503.2 Radiation Worker Training

Initial and annual requalification radiation worker training will be provided to personnel who require entry into the Radiologically Controlled Area (RCA). Training topics and scope will be commensurate with the stages of decommissioning, which typically will consist of information on the following topics:

- Fundamentals of radiation.
- Radiation and contamination measurement and control.
- Maintaining radiation dose as low as is reasonably achievable.
- Radioactive waste minimization.
- Radiation work permits.
- Radiation protection issues associated with decommissioning.

- Donning and removing protective clothing.

503.3 Certified Fuel Handler

YNPS has implemented, with NRC review and approval, a Certified Fuel Handler training program which replaced the 10CFR Part 55 NRC licensed operator training program (Reference 503-2). The

program includes provisions for training, proficiency testing, certification and recertification of the Certified Fuel Handler position. Certified Fuel Handlers are required by Technical Specifications to be on site at all times as part of the minimum shift crew composition, with additional staffing required for supervising fuel handling operations.

503.4 Specific Job Training

YNPS training programs will assure the following:

- Personnel responsible for performing activities are instructed as to the purpose, scope, and implementation of applicable controlling procedures.

- Personnel performing activities are trained as appropriate, in the principles and techniques of the activity being performed.

- The methods of implementing the training programs are documented.

503.5 Nonradiation Worker Indoctrination

Personnel who are not qualified radiation workers may be granted escorted access to the Radiologically Controlled Area (RCA). All personnel without radiation worker training require escorts that are qualified radiation workers.

503.6 Training Records

Training records will be maintained and retained in accordance with administrative plant procedures.

REFERENCES

503-1 BYR 93-055, Implementation of the Training Rule, 10CFR50.120, J. K. Thayer to M. B. Fairtile (USNRC), July 28, 1993.

503-2 NYR 92-122, NRC Approval of Certified Fuel Handler Program and Termination of Operator Licenses (TAC No. M83384), M. B. Fairtile (USNRC) to J. M. Grant, June 26, 1992.

503-3 NYR 93-149, NRC Partial Exemption from the Training Rule, 10CFR50.120 (TAC No. M87168), M. B. Fairtile (USNRC) to J. M. Grant (YAEC), November 19, 1993.

504 DEFUELED TECHNICAL SPECIFICATIONS

504.1 Description

After the decision to permanently cease YNPS reactor operations, YAEC submitted a proposed change in March 1992 to the NRC modifying the plant full power operating license to a possession only license status (Reference 504-1). The proposed change removed authority of YAEC to operate the reactor and move fuel back into either the Reactor Vessel or Vapor Container. The facility license was amended in August 1992 to a possession only status (Reference 504-2).

Coincident with the possession only license submittal, YAEC reviewed the plant licensing basis to determine the applicability of existing Technical Specifications to a permanently defueled condition. The Technical Specifications which were applicable to a defined mode associated with fuel in the reactor were determined to be not applicable to the permanently defueled condition. Technical Specifications that were determined to be applicable at all times were reviewed to assess their relevance to the permanently defueled condition. Individual Technical Specification changes were proposed for the higher priority items identified by the review. Table 504.1 summarizes the Technical Specification changes proposed by YAEC.

On December 23, 1992, a Technical Specification change was proposed to complete the transition of the Technical Specifications to be applicable to the permanently defueled condition (Reference 504-3). The proposed change eliminated specifications that were not applicable and reformatted the specifications that remained applicable. The NRC approved the license amendment on June 11, 1993 (Reference 504-4).

Limiting conditions for operation and surveillance requirements were reduced to the following items:

- <u>Applicability</u> - This specification describes the implementation of the limiting conditions for operation and surveillance requirements.

- <u>Spent Fuel Pit Water Level</u> - This specification presents a minimum spent fuel pit water level requirement to ensure the basis of the fuel handling accident and to minimize exposure to personnel during fuel handling.

- <u>Crane Travel Over Spent Fuel Pit</u> - This specification presents maximum load and load path restrictions for the spent fuel pit crane to protect fuel stored in the spent fuel pit.

- <u>Spent Fuel Storage Area Radiation Monitor</u> - This specification presents radiation monitoring operability requirements to ensure detection of inadvertent criticality during fuel handling.

- <u>Liquid Hold-Up Tanks</u> - This specification presents radioactive material curie limits for tanks that are not protected by engineered features that would contain inadvertent release of their contents.

- <u>Sealed Source Contamination</u> - This specification presents maximum contamination limits for sealed sources to ensure that allowable intake limits are not exceeded.

The design features and administrative controls were modified to delete portions that were not applicable to a permanently defueled condition.

No additional Technical Specification changes are needed to control decommissioning operations.

REFERENCES

504-1 BYR 92-037, Request For Modification of Yankee Nuclear Power Station's Operating License To Remove Authorization For Reactor Power Operation - Possession Only License, J. K. Thayer to T. Murley (USNRC), March 27, 1992.

504-2 NYR 92-148, Issuance of Amendment No. 142 To Facility License No. DPR-3 - Yankee Nuclear Power Station (TAC No. M83024), M. B. Fairtile (USNRC) to J. M. Grant, August 8, 1992.

504-3 BYR 92-112, Permanently Defueled Technical Specifications, J. K. Thayer to M. Fairtile (USNRC), December 21, 1992.

504-4 NYR 93-062, Issuance of Amendment No. 148 To Facility Possession Only License No. DPR-3 - Yankee Nuclear Power Station (TAC No. M85244), M. B. Fairtile (USNRC) to J. M. Grant, June 11, 1993.

TABLE 504.1

Technical Specification Changes for Transition to Permanently Defueled Condition

Submittal	Approval Date	Amendment Number	Reference
Minimum Shift Staffing	July 22, 1992	141	NYR 92-143, Issuance of Amendment No. 141 To Facility Operating License No. DPR-3 – Yankee Nuclear Power Station (TAC No. M83383).
Possession Only License	August 8, 1992	142	NYR 92-148, Issuance of Amendment No. 142 To Facility License No. DPR-3 - Yankee Nuclear Power Station (TAC No. M83024).
Transfer of Fire Protection Specifications to Manual	August 20, 1992	144	NYR 92-156, Issuance of Amendment No. 144 To Facility Possession Only License No. DPR-3 – Yankee Nuclear Power Station (TAC No. M83746).
PORC Administration Change	September 4, 1992	145	NYR 92-182, Issuance of Amendment No. 145 To Facility Possession Only License No. DPR-3 - Yankee Nuclear Power Station (TAC No. M84005).
RETS/REMP Transfer	November 5, 1992	146	NYR 92-183, Issuance of Amendment No. 146 To Facility Possession Only License No. DPR-3 – Yankee Nuclear Power Station (TAC No. M84372).
Permanently Defueled Technical Specifications	June 11, 1993	148	NYR 93-062, Issuance of Amendment No. 148 To Facility Possession Only License No. DPR-3 – Yankee Nuclear Power Station (TAC No. M85244).

SECTION 505

505 PROCEDURES

505.1 Normal Defueled Operating Procedures

The normal Operating Procedures used in the defueled condition have been developed in accordance with the Yankee Decommissioning Quality Assurance Program. The controlled procedures, utilized by plant department(s) personnel for the conduct of plant operations, are contained in the Plant Procedures Manual and are available for inspection.

505.2 Emergency Procedures

The Emergency procedures are comprised of the Emergency Operating Procedures and the Emergency Plan Implementing Procedures. These procedures are revised, reviewed and controlled in accordance with the Yankee Decommissioning Quality Assurance Program.

The Emergency Operating Procedures provide instructions to the Operations personnel. These procedures identify various plant system(s) and equipment failure/malfunction(s) that may occur, describe the conditions that enable operator recognition of such an event, and provide instructions that assist in mitigating an event and its consequence(s) to the plant, plant personnel, the public, and the environment.

The Emergency Plan Implementing Procedures (EPIPs) provide instruction to Emergency Response Organization (ERO) personnel. These procedures describe the responsibilities and provide for: Classifying emergencies; notifying plant personnel and off-site agencies; activating the Emergency Response Facilities (ERFs); controlling on-site radiation exposures; responding to medical emergencies, performing security activities; evaluating radiological data; formulating protective action recommendations; releasing public information; operating emergency computer equipment; and preparing for recovery of the plant. Other EPIPs provide for maintaining the ERFs, maintaining emergency response equipment, and training the ERO.

506 PREDICTIVE MAINTENANCE PROGRAM

The Predictive Maintenance Program is designed to ensure continued system reliability for those systems used to support Spent Fuel Pit cooling. The program is implemented through administrative and maintenance procedures. These procedures have been developed in accordance with the Yankee Decommissioning Quality Assurance Program. The controlled procedures are utilized for the conduct of maintenance activities and are available for inspection.

507 RADIATION PROTECTION PROGRAM

507.1 Introduction

The Code of Federal Regulations defines decommissioning as the activities necessary "to remove (as a facility) safely from service and reduce residual radioactivity to a level that permits release of the property for unrestricted use and termination of license" (10CFR50.2). A comprehensive Radiation Protection Program is needed to meet the objectives presented in the definition. YNPS intends to maintain essential elements of the Radiation Protection Program that were implemented successfully during its 31 year operating life to meet this goal. Changes to the program will be made as necessary to meet the needs of decommissioning.

The Radiation Protection Program has undergone and continues to undergo inspections and audits from both the NRC and the YAEC Quality Assurance Department. The purpose of these reviews is to ensure that the program complies with the Code of Federal Regulations, applicable regulatory guidance documents, and industry standards. YAEC is committed to maintaining a high level of performance and to enhancing the quality of the Radiation Protection Program throughout the YNPS decommissioning.

The YNPS Radiation Protection Program for decommissioning will continue to be implemented through existing YNPS administrative procedures. These procedures constitute the highest tier documentation of the Radiation Protection Program and define the radiation protection organization, responsibilities, authorities, administrative policies, program objectives, and standards to implement the Radiation Protection Program.

This section of the FSAR presents an overview of the Radiation Protection Program administrative and implementing procedures that will be used during decommissioning.

507.2 Management Policies

507.2.1 Management Policy Statement

YAEC is committed to the safe decommissioning of YNPS. The primary objective of the Radiation Protection Program is to minimize the actual and potential exposure of workers, visitors, and general public to radiation. YAEC and its contractors will provide sufficient qualified staff, facilities, and equipment to conduct a radiologically safe decommissioning. YAEC will continue to comply with regulatory requirements, radiation exposure limits, and radioactive material release limits. In addition, YAEC will make every effort to maintain radiation exposures and releases of radioactive materials in effluents to unrestricted areas as low as is reasonably achievable (ALARA). The ALARA philosophy will be incorporated into all decommissioning activities and will have full management support.

YAEC requires good radiation work practices as a condition of employment. Each radiation worker is responsible for performing work in a radiologically safe manner, consistent with the standards of conduct described in the Radiation Protection Program procedures.

This management policy will continue to be communicated to all radiation workers through General Employee Training and will continue to be incorporated into all applicable procedures.

507.2.2 Administrative Policy

YAEC will ensure that activities conducted during decommissioning will be managed by qualified individuals who will perform program operations in accordance with established procedures. Radiological hazards will be monitored and evaluated on a routine basis to maintain radiation exposures and the release of radioactive materials to unrestricted areas as far below specified limits as is reasonably achievable. Each element of the Radiation Protection Program will be defined and implemented using written procedures. Radiation protection training will be provided to all occupationally exposed individuals to ensure that they understand and accept their responsibility to follow procedures and to maintain their individual radiation dose as low as is reasonably achievable.

YAEC project management will ensure that work specifications, designs, and work packages involving potential radiation exposure or handling of radioactive materials incorporate effective radiological controls. Project supervisors will incorporate radiation protection considerations in the work activities under their control.

Radiation protection records will be prepared and maintained using high standards of accuracy, traceability, and legibility to meet the requirements of regulatory agencies and company procedures.

507.2.3 ALARA Policy

YAEC is committed to maintain an ALARA Program that is implemented based on guidance provided in Regulatory Guides 8.8 and 8.10 (References 507-2 and 507-3). All activities at YNPS involving radiation and radioactive materials will be conducted such that exposure of employees, contractors, and the general public to radiation is maintained as low as is reasonably achievable. This determination will consider the current state of technology and the economics of improvements in relation to their benefit (i.e., reduction of dose).

Appropriate ALARA considerations will be incorporated into decommissioning activity planning and design activities at an early stage to allow full consideration of reasonable alternatives. Final plant modifications also will be reviewed to ensure that ALARA was incorporated into the activities.

YNPS management will establish and monitor progress towards specific goals and objectives for the YNPS decommissioning ALARA Program.

507.2.4 Regulatory Compliance Policy

YAEC is committed to maintain the Radiation Protection Program in compliance with the requirements of the Code of Federal Regulations and, to the extent practical, information contained in industry standards, regulatory guides, and other guidance documents. The Radiation Protection Program is assessed against all new regulatory guidance and modified as necessary. YAEC implemented the revised 10CFR Part 20 on January 1, 1994.

507.2.5 Waste Minimization and Disposal Policy

YAEC will ensure appropriate processing, packaging and monitoring of solid, liquid and gaseous wastes during decommissioning by continuing to implement the Process Control Program, the Radiological Effluent Control Program and the Radiological Environmental Monitoring Program. These programs will be maintained in strict compliance with Technical Specification and Off-Site Dose Calculation Manual requirements to meet the requirements of 10CFR Parts 20, 50, 61, and 71; 49CFR; state regulations; disposal site requirements; and any other applicable requirements. Implementing procedures will be maintained for the classification, treatment, packaging, and shipment of radioactive material.

YNPS will continue to implement and enforce a Radioactive Waste Reduction Program to minimize the generation of radioactive wastes. YNPS management will monitor waste minimization efforts for decommissioning. All decommissioning personnel will receive training in the applicable procedures and practices to minimize the generation of radioactive waste.

507.2.6 Respiratory Protection Policy

YAEC is committed to minimizing the inhalation of air contaminated with dusts, mists, fumes, gases, vapors, and radionuclides at YNPS. The primary means of achieving this goal is to prevent or mitigate the hazardous condition at the source. Every reasonable effort will be made to achieve this objective by using engineering controls, including process modification, containment, and ventilation techniques. Respiratory protection equipment usage will be considered after engineering controls have been evaluated. Use of respiratory protection equipment for radionuclide inhalation reduction will be consistent with the goal of maintaining the total effective dose equivalent to personnel as low as is reasonably achievable.

The existing respiratory protection program will continue to be implemented and maintained in accordance with 10CFR Part 20 and other applicable regulatory guidance.

507.3 Decommissioning Exposure Projections

This information has been extracted, updated and relocated to the PSDAR within the FSAR.

507.4 ALARA Program

507.4.1 General Program Description

All activities at YNPS involving radiation and radioactive material will be conducted such that the radiation doses received by employees, contractors, and the general public are maintained as low as is reasonably achievable. This determination will consider the current state of technology and the economics of improvements in relation to the benefit (i.e., reduction of dose). The YNPS ALARA Program is implemented in an operating plant procedure, and is based on guidance provided in Regulatory Guides 8.8 and 8.10.

The following criteria will continue to be used to determine when an activity requires a specific ALARA review:

- The dose for the total completion of the activity exceeds 1 person-rem.
- The whole body dose rate field exceeds 5 R/hr for an activity other than a surveillance or inspection.
- The loose surface contamination exceeds 500,000 dpm/100cm2 (beta/gamma) for an activity other than surveillance or inspection.
- The airborne radioactivity concentration exceeds 40 DAC as a result of an activity or in the area of an activity.
- The work involves a planned special exposure.
- The Radiation protection staff or ALARA Coordinator request an ALARA review.

In addition to these criteria, ALARA considerations will be incorporated into decommissioning activity planning and design activities at an early stage to allow full consideration of reasonable alternatives. Plant modifications also will be reviewed to ensure that ALARA was incorporated into the activities. YNPS management will establish and monitor progress towards specific goals and objectives for the YNPS decommissioning ALARA Program.

507.4.2 ALARA Program Organization and Responsibilities

TheSafety OversightManager coordinates the ALARA Program scope and implementation. The ALARA Coordinator is responsible for completing the ALARA reviews. However, the actual implementation of specific ALARA actions, as incorporated into daily work activities, is the responsibility of each individual manager, supervisor, and worker.

An administrative plant procedure defines the responsibilities and authorities of the ALARA Committee. The primary responsibility of the ALARA Committee is to advise the Decommissioning Manager on matters related to exposure and contamination reduction. The committee will review the following:

- All plant decommissioning activities, maintenance activities, and modifications that have an estimated dose expenditure in excess of 10 person-rem.
- All ALARA post-job reviews for activities exceeding 10 person-rem.
- Any individual’s dose which exceeds the quarterly or annual administrative limits established for the individual.
- Any exposure or contamination issue of concern requested by committee members.

The ALARA Committee is chaired by the Decommissioning Manager. Other members of the committee include the Radiation Protection/Chemistry Manager, ALARA Coordinator, and other designated managers and supervisors involved in decommissioning activities. Members are given the appropriate authority and responsibility necessary to implement an effective ALARA Program.

507.4.3 ALARA Training and Instruction

Commitment to the principles of the ALARA Program will be reflected in all radiation protection training. Training courses will be evaluated by the Radiation Protection/Chemistry Manager to ensure that ALARA principles are incorporated into lesson plans.

507.5 Administrative Dose Control

Administrative radiation dose controls will continue to be implemented during decommissioning. Dose controls ensure the following:

- Personnel do not exceed regulatory dose limits.
- Equitable distribution of dose among available qualified workers.
- Collective dose to workers is as low as is reasonably achievable.

Administrative plant procedures implement the program to control and limit external and internal radiation exposure. These procedures include the following elements:

- A summary of administrative and regulatory dose limits.
- A description of Radiation Control Area Postings and Controls.
- A description of radiological survey data available on-site.
- Instructions on the use and care of dosimetry.
- Instructions on the conduct of work in the Radiation Control Area.
- Instructions on personal monitoring for contamination.

Personnel dose reports are prepared weekly with more frequent reports during periods of high work activity. The reports are distributed to each plant department and are posted in the Radiation Control Area Control Point. Decommissioning supervisors are responsible for reviewing the dose reports and planning high dose activities such that the dose is distributed as evenly as possible among available qualified personnel.

507.6 Radiation Work Permits

Radiation Work Permits will continue to be used to administratively control personnel entering or working in areas that have, or potentially have, radiological hazards present. The primary function of the Radiation Work Permit is to allow authorized activities to be conducted in radiologically controlled areas using safe and radiologically sound practices. The permit documents the work description, the worker names, the radiological conditions, and the radiological precautions and requirements. The permit also is an element of the ALARA program where it is used to screen activities to determine if a specific ALARA review is necessary and to track personnel and job exposure data.

An administrative plant procedure presents the requirements for requesting, using and terminating a Radiation Work Permit. An additional operating plant procedure presents the process used by the radiation protection staff to prepare, issue and monitor a Radiation Work Permit.

Radiation Work Permits are required for the following activities:

- Entry into a high or very high radiation area, an airborne radioactivity area, or any area posted with a sign stating that a Radiation Work Permit is required.

- All fuel handling operations.

- Maintenance on or inspections of equipment with loose surface contamination levels in excess of 10,000 dpm/100 cm2 (beta-gamma).

- When prudent radiation protection practices warrant the use of a Radiation Work Permit, as determined by the Radiation Protection/Chemistry Manager.

507.7 Area Definitions and Postings

An operating plant procedure describes the requirements for radiological postings at the entrance and boundaries of radiologically controlled areas. The purpose of the postings is to advise workers of radiological hazards that may be encountered in the areas. Informational postings may also be used to provide additional radiological instructions to workers. Each worker is responsible for the observance of the area postings and compliance with the indicated requirements.

507.8 External Dosimetry

507.8.1 General Considerations

External radiation dose will be monitored through the use of thermoluminescent dosimeters (TLD), direct reading dosimeters, and digital alarming dosimeters. The official record of external dose from beta and gamma radiation normally will be obtained from the TLD readings. Direct reading or digital alarming dosimeters will be used as a means for tracking dose between TLD processing and may also be used as a back-up to the TLDs. TLDs will be processed at a frequency that ensures personnel dose limits are not exceeded.

507.8.2 Monitoring Whole Body Dose

All decommissioning workers are required to wear external radiation monitoring devices whenever they enter the Radiation Control Area. Radiation workers are instructed to read the direct reading and digital alarming dosimeters prior to use and periodically during the work activity. The TLD and the direct reading or digital alarming dosimeters are worn typically in close proximity to each other on the trunk of the body between the neck and waist. Under certain conditions, where the chest or trunk may not be the location of highest whole body dose, dosimetry devices may be relocated.

Multiple whole body dosimetry may be used if work is to be performed in a nonuniform radiation field in which the dose to a portion of the body that is exposed to the highest dose source cannot easily be determined. In these cases, multiple sets of dosimeters will be worn on those portions of

the body expected to receive the highest dose. Guidance for conducting the evaluation and criteria for determining when multiple dosimetry is required is provided in an operating and departmental plant procedure.

Dosimetry requirements are specified on the Radiation Work Permit.

507.8.3 Dosimetry Quality Control

Periodic quality assurance checks of dosimetry will be conducted by exposing whole body, extremity and environmental dosimetry to known radiation doses and sending them to a qualified Environmental Laboratory for processing. A departmental plant procedure implements the quality control program for plant dosimetry. Discrepancies between the expected exposure and the laboratory results will be reconciled and documented. The Environmental Laboratory is accredited by the National Institute of Standards and Technology (NIST) under the National Voluntary Laboratory Accreditation Program (NVLAP) for dosimetry.

507.9 Internal Dosimetry Control and Monitoring

507.9.1 General Considerations

Internal radiation dose inherently is more difficult to measure than external radiation dose, but it is generally much easier to prevent. Therefore, major emphasis is placed on preventing internal radiation dose, as long as it is consistent with the goal of keeping total effective dose as low as is reasonably achievable.

The primary methods for controlling the intake of radioactive material into the body is identifying and minimizing the sources of airborne radioactivity and applying engineering controls to reduce airborne radioactivity concentrations. The use of respiratory protection will be used after the primary methods have been implemented to the extent practicable.

An administrative plant procedure describes the program that is implemented to monitor potential internal radiation exposures.

507.9.2 Bioassay Program

Whole body counting (in vivo) is the primary method that is used to determine the identity and quantity of gamma emitting radionuclides present in the body. Radiation workers will receive, as a minimum, an initial, an annual, and a termination whole body count. In addition, personnel will receive a whole body count after a suspected intake of radioactive materials. Radiation protection implementing procedures provide guidance on whole body counter operation, calibration, and quality control.

Indirect bioassay (in vitro) measurements will be made, as necessary, to monitor for alpha and beta emitting radionuclides and to provide data for calculation/determination of internal dose. This method of bioassay will typically be used only for radionuclides which cannot be determined by whole body counting or when additional information on an intake is required. Radiation protection implementing procedures include criteria for indirect bioassay and methods for data analysis and interpretation.

An administrative plant procedure is used to implement the bioassay program at YNPS.

507.10 Respiratory Protection Program

A Respiratory Protection Program will continue to be maintained in accordance with 10CFR Part 20 and other applicable regulatory guidance. The primary means of providing respiratory protection is to prevent or mitigate the hazardous condition at the source. Every reasonable effort will be made to achieve this objective by using engineering controls, including process modification, containment, and ventilation techniques. Respiratory protection equipment usage will be considered after engineering controls have been evaluated. Use of respiratory protection equipment will be consistent with the goal of maintaining the total effective dose to personnel as low as is reasonably achievable.

An administrative plant procedure is used to implement the Radiological Respiratory Protection Program at YNPS.

507.11 Radioactive Material Controls

The Radiation Protection Program establishes radioactive material controls that ensure the following:

- Prevention of inadvertent radioactive material release to uncontrolled areas.
- Assurance that personnel are not exposed inadvertently to radiation from radioactive materials.
- Minimization of the amount of radioactive waste material generated during decommissioning.

Radioactive material is defined as any of the following: 1) material activated by YNPS reactor operation, 2) material contaminated from the operation or decommissioning of YNPS, or 3) licensed material procured and used to support the operation and decommissioning of YNPS.

All materials leaving the Radiation Control Area, and subsequently the YNPS site, will be surveyed to ensure that radioactive materials are not inadvertently discharged from the facility. An administrative plant procedure will be used to ensure that all potentially radioactive or contaminated items removed from the Radiation Control Area or the YNPS site are surveyed. This procedure was written to incorporate the guidance presented in NRC Circular No. 81-07 and NRC Information Notice No. 85-92 (References 507-5 and 507-6). The following survey methods (instrumentation is typical) will be used:

- <u>Materials and Equipment</u> - Direct frisking with a portable Geiger-Mueller, gas proportional and scintillation detectors (as appropriate). Large quantities of materials with a low probability of being contaminated (e.g., secondary side plant components) may be released through the analysis of representative samples (as appropriate for the materials being monitored) combined with aggregated waste surveys.

- <u>Smear Samples</u> - Analysis with a Geiger-Mueller, gas proportional, scintillation, and solid state detector (as appropriate).

- Bulk Liquids or Solids - Analysis with high resolution gamma spectrometry system to the environmental lower limit of detection. Large quantities of materials (e.g., excavated soils) may be released through the analysis of representative samples combined with surveys of the aggregated wastes (e.g., dumpsters).

Materials will be released if no discernable plant-related activity is detected within the capability of the survey methods presented above. Any radioactive material that is shipped from the site is handled in accordance with an operating plant procedure which ensures compliance with NRC and Department of Transportation requirements.

An administrative plant procedure provides instructions regarding the proper handling and storage of contaminated tools and equipment. This procedure ensures that tools and equipment are decontaminated promptly. Tools and equipment that are not fully decontaminated are stored in designated radioactive material storage areas. An operating plant procedure ensures that the areas where radioactive materials are stored are posted clearly.

507.12 Surveillance

Routine radiological surveillances will continue to be conducted during decommissioning to monitor radiation sources, to determine radiological conditions, and to comply with the requirements of 10CFR Part 20. Surveys also will be performed to evaluate radiological conditions in support of decommissioning work activities. Operating plant procedures will be used to implement the surveys. These procedures specify the types of instrumentation, survey methods, and review requirements for each survey performed.

The final radiation survey that will be completed following decontamination and dismantlement activities is described in Section 307.

507.13 Instrumentation

A sufficient inventory and variety of operable and calibrated portable, semi-portable, and fixed radiological instrumentation will be maintained on site to allow for effective measurement and control of radiation exposure and radioactive material and to provide back-up capability for inoperable equipment. Equipment will be capable of measuring the range of gamma, beta, alpha and neutron dose rates and radioactivity concentrations expected. Instrumentation will be calibrated at prescribed intervals or prior to use against certified equipment having known valid relationships to nationally recognized standards. An administrative plant procedure will be used to control the use of radiation protection instrumentation.

Installed process and effluent monitors are used in accordance with the Off-Site Dose Calculation Manual (Reference 507-7).

507.14 Review and Audit

To ensure the Radiation Protection Program is effectively implemented and maintained, an organized system of reviews and audits will continue to be implemented during decommissioning in accordance with the Quality Assurance Program presented in Section 514.

507.15 Radiation Protection Program Performance Analysis

An administrative plant procedure will be used during decommissioning to evaluate the causes of unacceptable performance, to initiate corrective actions, and to trend overall performance. This process will be used to address the following types of deficiencies:

- Work activities generating unnecessary radiation exposure or contamination.
- Procedural actions resulting in unacceptable radiological performance.
- Unacceptable radiological work practices resulting in personnel contamination, spread of contamination, or unnecessary radiation exposure.
- Activities resulting in unnecessary generation of liquid or solid radioactive waste.
- Activities violating Radiation Work Permit instructions, postings, signs, and radiation protection implementing procedures.

Incidents that result in more serious radiological events will be reported using the same administrative plant procedure. These events include overexposures, large intakes of radioactive material, unplanned radioactive releases and significant radioactive spills. This procedure ensures that immediate and written notifications are made in accordance with regulatory requirements.

REFERENCES

507-1 NYR 91-097, Systematic Assessment of Licensee Performance (SALP) Final Report for Yankee Nuclear Power Station for the Period August 1, 1989 to January 15, 1991 (50-29/89-99), T. T. Martin (USNRC) to A. C. Kadak, May 20, 1991.

507-2 Regulatory Guide 8.8, Information Relevant to Ensuring That Occupational Radiation Exposure at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable.

507-3 Regulatory Guide 8.10, Operating Philosophy For Maintaining Occupational Radiation Exposure As Low As Is Reasonably Achievable.

507-4 Regulatory Guide 1.8, Personnel Selection and Training.

507-5 NRC IE Circular No. 81-07: "Control of Radioactively Contaminated Material," May 14, 1981.

507-6 NRC IE Information Notice No. 85-92: "Surveys of Wastes Before Disposal From Nuclear Reactor Facilities," December 2, 1985.

507-7 Off-Site Dose Calculation Manual.

507-8 CRP ALARA Committee Meeting 93-3 Agenda Memo, B. Cox to G. Babineau, August 28, 1993.

508 RADIOACTIVE WASTE MANAGEMENT

YNPS decommissioning requires handling of a large volume of radioactive materials to reduce residual radioactivity to a level permitting release of the site for unrestricted use and termination of license. Materials that are not decontaminated and released will be processed as radioactive waste. This section of the FSAR presents the programs used to manage spent fuel and control the processing of solid, liquid and gaseous radioactive waste.

YAEC will continue to ensure appropriate processing, packaging and monitoring of solid, liquid and gaseous wastes during decommissioning by implementing the Radiation Protection procedures, the Process Control Program, the Radioactive Effluent Controls Program and the Radiological Environmental Monitoring Program. These programs will be maintained in compliance with Technical Specification requirements to meet federal and state regulations, disposal site requirements, and any other applicable requirements. The YNPS radioactive waste management program is implemented through the Radiation Protection Program (Section 507). Implementing procedures will be used to control the classification, treatment, packaging and shipment of radioactive material.

508.1 Solid Radioactive Waste Processing

Solid radioactive waste handling at YNPS is divided into three phases: packaging, on-site storage awaiting shipment and shipment. Each of these phases will be implemented in strict compliance with Technical Specifications and applicable federal, state and disposal site requirements. In addition, all waste processing activities will be completed in accordance with the requirements of the Decommissioning Quality Assurance Program (Section 514).

YNPS decommissioning solid waste is comprised of both high and low level radioactive waste. A portion of the reactor vessel internal components (e.g., core baffle) have radionuclide concentrations in excess of the 10CFR Part 61 Class C limits. These materials are not generally acceptable for near-surface disposal and have been classified as high level radioactive waste. High level radioactive waste will be stored with the fuel and will be shipped to an appropriate disposal facility after a location becomes available.

508.1.1 Solid Radioactive Waste Packaging

Radioactive waste packaging at YNPS will be performed in areas that minimize radiation exposure to personnel, control the spread of contamination, and are adequate for packaging activities. Examples of potential on-site waste packaging areas are: Compactor Building and Potentially Contaminated Area Warehouse, Vapor Container, Primary Auxiliary Building, Spent Fuel Pit Building and FTE - Fuel Transfer Enclosure. Temporary facilities, that meet the requirements above, also may be constructed for waste packaging.

Radioactive waste packaging operations will be implemented through plant procedures that ensure the following:

- A Radiation Work Permit has been issued for handling radioactive materials.
- Specific packaging requirements are identified.

- Quality assurance personnel have been notified of packaging operations.
- Containers are surveyed for external contamination.

Personnel will conduct inspection activities in accordance with the requirements of the Decommissioning Quality Assurance Program (Section 514).

Waste packages will meet the requirements for transportation and disposal for each decommissioning waste stream. Examples of the waste containers that may be used are drums, boxes, liners, high integrity containers, sea-land containers, shielded casks, and other specialty containers. Waste container selection will be determined by the size, weight, classification, and activity level of the material to be packaged. In all cases, packaging will comply with requirements specified by 49CFR, 10CFR Part 71, and the disposal facility site criteria, as applicable.

A plant procedure provides instructions for determining the 10CFR Part 61 classification of low level radioactive waste. The procedure is used to determine the radionuclide content of a container through a combination of direct measurements and radiation shielding calculations.

Spent resins, filter media and other wetted wastes requiring stabilization will be processed in accordance with the Process Control Program (Reference 508-4), which is implemented through plant operating procedures. Whenever possible, stabilization will be completed inside the disposal package or liner to minimize additional waste handling prior to disposal.

508.1.2 Solid Radioactive Waste Storage Awaiting Shipment

Solid radioactive waste awaiting shipment to a disposal facility normally will be stored in the following locations:

- Potentially Contaminated Area Storage Building No. 1
- Potentially Contaminated Area Warehouse
- On-Site Storage Casks (e.g., OSSC)

Supplemental shielding may be used to shield waste containers to assure that exposure rates are maintained as low as is reasonably achievable. The resulting direct off-site dose in a year from storage of low-level radioactive waste in the PCA Warehouse and the PCA Storage Building No. 1 will be less than 1 mrem. A total radioactivity limit will be established for the storage locations to limit the off-site consequence of an event that dislodges supplemental shielding.

Stored waste will be inspected to assure that container integrity is maintained. The inspections will include visual inspections of a representative sample of containers for container integrity.

Radioactive materials will be stored such that in the event of a fire or an explosion, sufficient fire detection, response, and suppression capability in conjunction with spatial separation will ensure that any radiological release would be bounded by the accident analyses presented in Section 400.

Large components awaiting shipment may be stored in the Yard Area prior to shipment. Precautions will be taken to ensure that the components are within barriers, as necessary, and adequately protected from on-site hazards (e.g., heavy load movement).

508.1.3 Solid Radioactive Waste Shipment

Solid radioactive wastes will be shipped in compliance with applicable federal and state regulations. Operating plant procedures present the requirements for radioactive materials shipment. These procedures ensure the following:

- Appropriate labeling and documentation of shipping containers.
- Quality assurance oversight of shipment preparation.
- Verification of acceptable package physical condition and contamination levels.
- Appropriate permits and licenses for waste shipment.
- Notification of appropriate governmental agencies prior to shipment.

Most of the radioactive material and waste shipments will be completed over roads. Rail transportation may be used for heavy shipments (e.g., Reactor Vessel). The routing of shipments may vary with weather and highway conditions. Additionally, local and state restrictions pertaining to radioactive material transport may affect some route selections. The carrier is responsible for selecting the appropriate route, which must conform to applicable federal, state, and local shipping requirements and be in accordance with Department of Transportation and NRC regulations.

508.2 Liquid Radioactive Waste Processing

Contaminated water will be generated during YNPS decommissioning as a result of draining, decontamination and cutting processes. The contaminated liquids will be processed either in the liquid waste evaporator or in a temporary facility (e.g., ion exchange and filtration system, solidification system). All liquid radioactive waste will be processed in accordance with the Process Control Program, the Off-Site Dose Calculation Manual (ODCM), Yankee Decommissioning Quality Assurance Program (YDQAP), applicable Technical Specifications and plant procedures.

The Process Control Program presents the administrative and technical controls for the liquid radioactive waste system to assure that waste meets shipment and disposal facility requirements. Liquid waste processing is monitored to assure safe operation, storage and disposal of waste to approved waste disposal sites. Liquids released from the site are monitored and controlled to ensure all releases of radioactivity to the environment are as low as is reasonably achievable. The Process Control Program is maintained in accordance with Technical Specification 6.12.

The YDQAP establishes two programs affecting radioactive liquids processing: Radioactive Effluent Controls Program, Radiological Environmental Monitoring Program. The Radioactive Effluent Controls Program conforms with 10CFR50.36a requirements to control radioactive effluents and to maintain dose to members of the public from radioactive effluents as low as is reasonably achievable. This program is presented in the ODCM (Reference 508-5) and implemented through several plant procedures. This program complies with the requirements of Technical Specification 6.13.1 and the YDQAP.

The ODCM contains methodologies and parameters used in the following:

- Calculation of off-site doses resulting from radioactive gaseous and liquid effluents.
- Calculation of gaseous and liquid effluent monitoring alarm and trip setpoints.
- Conduct of the Radiological Environmental Monitoring Program.

The ODCM forms the basis of plant procedures which document the off-site doses due to plant operation. The off-site dose calculations demonstrate compliance with the numerical guides for design controls of 10CFR Part 50, Appendix I. Several plant procedures implement the ODCM requirements.

508.3 Airborne Radioactive Waste Processing

Airborne radioactive waste processing is limited to radioactive particulate emissions during decontamination and dismantlement activities. Exhaust air from various plant buildings, as described in Section 231, Ventilation System, is filtered through a high efficiency filter assembly before discharging to the Primary Vent Stack. Instrumentation channels monitor gas released through the Primary Vent Stack. An operating plant procedure ensures that airborne releases are monitored and maintained within the limits of the ODCM.

Dismantlement activities will be designed to ensure that airborne releases are monitored to the maximum extent practicable by implementing the following considerations during detailed planning of decommissioning activities:

- The VC Ventilation and Purge System will be maintained in operation during decontamination and dismantlement activities in the Vapor Container when a significant radioactive source term is present.

- The Ventilation System will be maintained in operation during decontamination and dismantlement activities in the Primary Auxiliary Building (PAB) cubicle area, Spent Fuel Pit Building, and Waste Disposal Building when a significant radioactive source term is present.

- Local HEPA filtration systems will be used when activities could result in the release of significant radioactive particulates. The local HEPA filtration systems should exhaust to areas served by the Ventilation System when used outside of the Vapor Container to monitor particulate releases. If the work area is a significant distance from the plant Ventilation System, it may not be practical to meet this requirement. In these cases, monitoring of the HEPA Filtration System exhaust will be performed. Prior to initial deployment and periodically, each HEPA System is tested against Dioctyl Phthalate (DOP) with an acceptance criterion of 99.95% filter retention. This ensures that airborne particulate releases to the environment have been reduced to as low as is reasonably achievable.

Airborne effluents from the Primary Vent Stack will be monitored and be reported using installed plant equipment and established procedures in accordance with Off-Site Dose Calculation Manual requirements. Local supplemental air monitoring will be performed to support decommissioning activities.

508.4 Mixed Waste

YAEC has a nonradioactive waste management program to ensure compliance with all the federal and state hazardous waste regulatory requirements. The use of hazardous materials and the generation of hazardous wastes are controlled through the nonradioactive hazardous waste management program. This program is presented in Section 511.

YNPS currently has a quantity of mixed waste, including Polychlorinated Biphenyls (PCBs) contaminated material. There are varying quantities of oil, trichloroethane, rust remover, degreasers, chromated evaporator bottoms, decontamination and scintillation solutions.

No chemicals or other substances are anticipated to be used during decommissioning operations that could become mixed waste. If mixed wastes are generated, they will be managed according to Subtitle C of the Resource Conservation and Recovery Act (RCRA) to the extent it is not inconsistent with NRC handling, storage, and transportation regulations.

Mixed wastes from the YNPS will be transported only by authorized and licensed transporters and shipped only to authorized and licensed facilities. If technology, resources, and approved processes are available, processes will be evaluated to render the mixed waste nonhazardous.

YAEC has identified the presence of solid PCBs in some paint coatings, primarily in the Radiation Control Area. As in the cases of radiologically contaminated lead paint, asbestos, and other hazardous materials, contaminated paint that contains PCBs will be managed according to all applicable federal and state regulations.

508.5 Radioactive Waste Minimization

508.5.1 Radioactive Waste Reduction Program

YNPS will continue to implement and enforce a Radioactive Waste Reduction Program through implementation of an administrative plant procedure. YNPS management will monitor waste minimization practices for decommissioning. All decommissioning personnel will receive training in the applicable procedures and practices to minimize the generation of radioactive waste.

All workers entering the Radiation Control Area will receive radiation worker training. This training will include a review of work techniques that prevent unnecessary contamination of areas and equipment, practices for reuse of materials, and policies to prevent the unnecessary generation of mixed or radioactive wastes.

The following radioactive waste volume reduction methods will be incorporated into YNPS decommissioning activities:

- <u>Prevention of Waste</u> - Unnecessary generation of radioactive wastes will be controlled by procedures established to prevent unnecessary packaging, tools, and equipment from entering Radiation Control Area.

- <u>Decontamination and Re-Use of Materials</u> - Materials will be reused to the maximum extent practicable. Typical materials reused during the decommissioning include contaminated tools,

equipment, and clothing. Contaminated tools and equipment storage areas will be maintained. Protective clothing will be laundered and made available for reuse.

Voids in disposal containers will be filled with other contaminated material to reduce the total volume of waste for disposal to the maximum extent practicable. This produces a better waste form, maximizing burial efficiency, and minimizing project cost, disposal site usage, and transportation risk.

508.5.2 On-Site Decontamination Methods

On-site decontamination techniques will be used for processing and volume reduction of radioactive materials. The following are examples of decontamination methods that may be used during decommissioning:

- Strippable Coatings - Strippable coatings may be used to lift radionuclides from contaminated surfaces. A strippable coating is typically applied in a manner similar to spray painting a surface. Additives in the coating are designed to attract and to combine chemically with radioactive contaminants. Once the coating is dry, the contaminant is locked in the coating. The dried coating is easily removed from the surface, stripping the film containing the contamination. The stripped film is packaged and processed as a solid waste. Strippable coating may also be used to protect surfaces from becoming contaminated.

- Chemical or Solvent Decontamination - Chemical and solvent decontamination methods remove contamination by creating a solution of the radionuclides and the solvent used. This type of decontamination may be difficult to control because of the aggressiveness of the chemicals and solvents. However, the systems decontaminated at YNPS will not be returned to service after decontamination, therefore, excessive metal wastage is not significant. Chemicals used for decontamination will be evaluated for hazardous constituents. If the chemical could become a listed or characteristic hazardous mixed waste it will not be used.

- Dry Abrasive Impingement - Dry abrasive impingement is effective for removing heavy or tightly adhering oxide films. Examples of this technology are sandblasting and dry ice blasting.

- Water Washing - High pressure water washing is effective for removing surface contamination and for sluicing sludge from tanks. Barriers must be established to ensure that wash water is collected and processed in the plant liquid waste processing system.

- Vacuum Cleaning - HEPA filtered vacuum cleaners may be used in areas of high loose surface contamination.

508.5.3 Off-Site Radioactive Materials Processing

Several off-site radioactive materials processing options are currently available. The current decommissioning cost estimate (See PSDAR Section) assumes that the availability of processing alternatives is limited and that all significantly contaminated and activated materials are sent to a low level radioactive waste disposal facility. However, all processing alternatives will be evaluated during decommissioning to determine the most effective processing of radioactive materials.

The following are examples of off-site processing alternatives for radioactive materials removed during YNPS decommissioning:

- Decontamination - Decontamination facilities provide a wide range of decontamination technologies at centralized locations. The variety allows selection of appropriate technologies for each component of the decommissioning waste stream.

- Volume Reduction - Volume reduction facilities provide various processes (e.g., sorting, super-compaction) to reduce the volume of material that is sent to the disposal facility. This processing alternative is attractive for asbestos compaction which requires specialized containment during processing.

- Incineration - Incineration facilities safely incinerate materials resulting in very high volume reduction rates. Appropriate materials may include paper, certain plastics, lubricating oils, and solvents.

- Metal Melting - Metal melting materials process low specific activity metals. The processed metal is recycled to the nuclear industry as shielding and potentially in the future as cask liners and fuel canisters.

Waste packages will be transported to off-site facilities primarily in sea-land containers selected to meet transportation and receipt requirements of the off-site processing facility. Voids in transport containers are not a significant concern. However, efficient management of transportation resources is an important consideration to minimize the total number of shipments and decommissioning costs.

Radioactive material control and accountability procedures will be implemented to track material originating from YNPS during receipt, sorting, processing, and packaging for disposal. Off-site processing facilities will be selected that provide adequate radioactive material control and accountability procedures.

508.6 Decommissioning Radioactive Waste Projections

This information has been extracted, updated and relocated to the PSDAR within the FSAR.

REFERENCES

508-1 58-FR-34947, Notification of Spent Fuel Management and Funding Plans By Licensees of Prematurely Shut Down Power Reactors, June 30, 1993.

508-2 YRP 435/92, Spent Nuclear Fuel Storage Study Report and Recommendations, B. W. Holmgren, J. M. Buchheit, R. A. Mellor to J. K. Thayer, October 9, 1992.

508-3 YRP 303/93, Impact of Wet Spent Fuel Storage on Decommissioning, P. A. Rainey to R. A. Mellor, July 15, 1993.

508-4 Yankee Nuclear Power Station Process Control Program.

508-5 Yankee Nuclear Power Station Off-site Dose Calculation Manual.

509 TESTS

Plant tests are performed to ensure the continuous, safe, and efficient operation of equipment necessary to support the defueled condition and spent fuel pit cooling. The various types of tests are those conducted in accordance with either Technical Specification mandated requirements, governing applicable engineering practices and standards (ASME, IEEE, ISA, etc.), or those based on good engineering judgement and operational experience. The testing pertains principally to systems necessary to support spent fuel pit cooling.

Tests associated with systems necessary to support spent fuel pit cooling include but are not limited to the following:

- Plant Area Radiation Monitoring
- Site Environmental Radiological Surveillance
- Spent Fuel Pit Cooling System Performance

510 OCCUPATIONAL SAFETY PROGRAM

510.1 Introduction

This section provides an overview of the YAEC Occupational Safety Program as provided in the site approved Safety Manual, applicable plant procedures, and approved contractor programs.

510.2 Management Policy Statement

YAEC and its management are committed to the safe decommissioning of YNPS. The primary objective of the Occupational Safety Program is to protect workers and visitors from industrial hazards that have the potential of developing during decommissioning activities. YAEC and its contractors will provide sufficient qualified staff, facilities, and equipment to perform decommissioning in a safe and effective manner. YAEC is committed to compliance with federal and state requirements and to the guidance provided through industry standards and good work practices.

510.3 Health and Safety Organization and Functions

The existing Occupational Safety Program provides the basis for controlling safety during decommissioning activities. The purpose of the health and safety organization is to ensure that the standards of safety are maintained through effective implementation of the Occupational Safety Program. The effective implementation of the Occupational Safety Program is the responsibility of all decommissioning personnel:

- <u>Decommissioning Site Manager</u> - The Decommissioning Manager has the overall responsibility for safe operation and decommissioning activities of the plant and has control over those on-site resources necessary to meet this objective (TS 6.2.1.c). Included in this is the responsibility for assuring effective implementation of the Occupational Safety Program and assuring that all organizations involved with decommissioning are coordinated to achieve the goals of providing a safe work place and the reduction of industrial hazards.

- <u>Safety Oversight Manager</u> - The Safety Oversight Manager is responsible for the development and implementation of the Occupational Safety Program policies and standards. The Safety Oversight Manager has the authority to cease any work activity when worker safety is jeopardized or an unsafe condition occurs.

- <u>Health and Safety Staff</u> - Health and Safety staff report to the Safety Oversight Manager and are responsible for the day-to-day communication of safety. Health and Safety staff have the authority to cease any work activity when worker safety is jeopardized or an unsafe condition occurs.

- <u>Decommissioning Supervisors</u> - All supervisory personnel are responsible for the supervision and direction of safety practices during decommissioning activities.

- <u>Decommissioning Workers</u> - All plant and decommissioning workers are responsible for their own safe work practices as presented in the Safety Manual and OSHA Standards.

510.4 YAEC Occupational Safety Program

The YAEC Occupational Safety Program was developed to establish and maintain a safe work place for YAEC workers, contractors, and visitors. The program provides guidelines and procedures to be used to reduce industrial hazards and risks.

The site approved Safety Manual provides guidelines and requirements which will be incorporated into the detailed decommissioning planning process.

The following areas are discussed in the manual:

- Personnel Protection and Safety Equipment
- Prevention of Falls
- Safe Use of Ladders and Scaffolding
- Safe Handling of Hazardous Substances and Materials
- Safe Use of Hand and Portable Powered Tools and Equipment
- Welding, Cutting, and Brazing Safety
- Electrical Safety
- Confined Space Safety Requirements
- Heat Stress Prevention

Additional safety guidelines and instructions are included in plant procedures which receive a Health and Safety Organization review during the approval process. Health and Safety will review design change processes prior to commencement of work activities (Section 200.3.2).

510.5 Safety Training and Meetings

Safety training is conducted as part of the General Employee Training process, during routine safety meetings, and for job-specific purposes. The safety meetings focus on current safety issues and events as well as providing a forum for workers to ask questions and provide feedback.

511 NONRADIOACTIVE WASTE MANAGEMENT

511.1 Introduction

This section provides an overview of the YAEC Nonradioactive Waste Management Program and applicable plant procedures.

Site materials that are routinely handled as hazardous waste during disposal include solvents, oils, and absorbent materials used with these items. Nonroutine wastes such as paint containing Polychlorinated Biphenyls (PCBs), mercury, paint, and battery acid are also handled and disposed of through this program.

511.2 Management Policy Statement

YAEC and its management are committed to the safe decommissioning of YNPS. The primary objective of the YAEC Nonradioactive Waste Management Program is to protect workers, visitors, and the environment from the potential effects of hazardous materials. YAEC is committed to strict compliance with all federal and state hazardous waste handling and disposal requirements.

511.3 Hazardous Material Management

YAEC is required by the OSHA Hazard Communication Standard (29CFR1910.1200) to provide information to its employees and contractors concerning the hazardous substances to which they may be exposed. Administrative plant procedures were implemented to meet these requirements. General Employee Training was revised to apprise employees and contractors of hazardous materials used at YNPS.

511.4 Hazardous Waste Management

The YAEC Nonradioactive Waste Management Program was established to assure compliance with all the federal and state hazardous waste regulatory requirements. Nonradioactive hazardous wastes from YNPS are transported only by authorized and licensed transporters and shipped only to authorized and licensed facilities.

The hazardous waste management program is implemented through an administrative plant procedure. This procedure provides direction for the handling, temporary storage and preparation for shipment of nonradioactive hazardous waste. Routine preventive and emergency response procedures have been developed for precluding and containing hazardous material incidents.

511.4.1 Above-Ground Fuel Oil Storage Tanks

There are five above-ground oil storage tanks at YNPS:the Diesel Fire Pump fuel oil tank (TK-56), the two Security Diesel Generator fuel oil tanks (TK-76-1 and TK-76-2), and the two fuel oil tanks (TK-83-1 and TK-83-2) for the Fire Water Storage Tank Heating System. Each has a capacity of 275 gallons. In addition, there is a 500-gallon mobile fuel tank and a 750-gallon stationary tank that is part of a skid-mounted, standby diesel generator.

These tanks will remain in service throughout the dismantlement phase. When no longer required, the tanks will be emptied, cleaned and disposed of by an authorized and licensed contractor.

511.4.2 Underground Fuel Oil Storage Tanks

Two underground fuel oil storage tanks supplied fuel oil to the Safe Shutdown System Emergency Diesel Generator and the Security Diesel Generator. The tanks have been excavated, removed and disposed of by an authorized and licensed contractor.

511.4.3 Underground Waste Oil Tank

The Underground Waste Oil Tank was used to store uncontaminated waste oil prior to shipment to a licensed disposal facility. The tank has been excavated, removed and disposed of by a licensed contractor.Uncontaminated waste oil is and will continue to be stored in drums in the Hazardous Waste Storage Area until disposal.

511.4.4 PCB Contaminated Transformer Oil

PCBs were present in the transformer oil in the Nos. 4, 5 and 6 Station Service Transformers. Each transformer contained approximately 370 gallons of oil. The transformers and oil have been disposed of by an authorized and licensed contractor.

511.4.5 Mercury

Mercury contained in instruments and switches will be removed and collected prior to final disposal of the equipment. The mercury will be reclaimed or processed by an authorized and licensed contractor.

511.4.6 Asbestos Containing Materials

Asbestos Containing Material (ACM) has been identified on many plant systems and in most areas and buildings. Asbestos insulating material was replaced with non-asbestos material as a part of maintenance activities that required insulation removal and was labeled accordingly. Decommissioning activities have removed most of the systems originally covered with asbestos insulating material. Minor quantities of non-insulation asbestos containing materials, in the form of gaskets and packing, remain present in numerous systems at the plant. In addition, corrigated metal siding on the Turbine Building and VC elevator contain asbestos. A licensed contractor has performed an asbestos survey of all remaining and accessible building materials.

Asbestos material will be removed by a licensed contractor prior to the start of dismantlement activities. Insulating material will be considered to be asbestos material unless marked “NON-ASBESTOS”. All asbestos containing materials will be removed and processed in accordance with an administrative plant procedure to ensure compliance with federal and state regulations.

Radiologically contaminated asbestos containing material and non-asbestos material will be disposed of in accordance with the requirements given in Section 508.

511.4.7 Lead-Based and PCB-Containing Paints

Lead-based and PCB-containing paints were used at YNPS to coat many steel components, concrete structures, and carbon steel piping. During the operating life of the paint, some of these paints have been covered with several coats of non-lead or non-PCB based paint.

Controls of the lead-based paint identification and removal process have been implemented to control workers exposure and ensure proper handling of lead materials. Lead-based paints on non-recycleable components (e.g., concrete) are removed, processed, and disposed of by qualified personnel.

Controls of the PCB-containing paint identification, removal, and disposal process have also been implemented. The controls are part of plant programs to manage workers exposure and ensure proper handling of PCB waste. An Alternative Method of Disposal Approval issued by the U.S. Environmental Protection Agency authorizes management of a specific category of PCB wastes designated as “PCB Bulk Product” wastes.

Work associated with both lead and PCB paints are controlled in accordance with administrative plant procedures.

511.5 Sampling and Remedial Actions

Site structures and environs are evaluated for non-radioactive hazardous materials by appropriate sampling and analytical protocols. Remedial actions are implemented as necessary to meet all federal, state, and local environmental quality requirements.

511.6 Industrial Waste Management

The final dismantlement of YNPS will require the handling and disposal of system and building wastes. These wastes will include non-hazardous materials that were never radiologically contaminated, have been decontaminated to meet release criteria, or material that contain asbestos or other hazardous materials. Non-radioactive non-hazardous wastes are expected to include the following:

- System Piping and Components (e.g., pumps, valves, tanks, nonasbestos insulation, heat exchanges, and supports).
- Duct-Work and Associated Equipment (e.g., duct, fans, filters, and supports).
- Electrical Systems and Equipment (e.g., cables and trays, conduit, motor control centers, generators, motors, and panels).
- Buildings and Structures (e.g., concrete, structural steel, roofing materials, siding, doors, and windows).

The materials presented above will be processed in accordance with the rules and regulations governing the disposal of nonradioactive, nonhazardous wastes.

511.7 Training

All personnel involved in the handling of hazardous materials and wastes receive initial and annual training. This training includes information on the types of hazardous materials and wastes, handling precautions, temporary storage locations, and emergency response procedures.

512 SECURITY PLAN

512.1 General

After the decision to permanently cease YNPS power operation, YAEC completed a comprehensive review of the security requirements for YNPS. The review incorporated the following:

- Re-evaluation of the design basis threat matrix to determine security needs for a permanently defueled condition.

- Evaluation of the Vital and Access Control Areas to determine the appropriate level of protection for the permanently defueled condition. This analysis reviewed existing safety analyses, the potential consequences of radiological sabotage events and the potential consequences of fuel storage events. The evaluation indicated that nuclear security could be focussed on spent fuel storage.

- Physical survey of the site to determine physical modifications needed to implement a reduced protected zone. Implementation of the recommendations of this survey allowed reduction of the protected area and the implementation of an industrial security zone for most of the plant site.

Based on this evaluation, the security plan was modified creating a YNPS Defueled Security Plan (References 512-1 and 512-2). The NRC approved the plan and on December 18, 1992, the plan was implemented by YNPS (Reference 512-3). The Defueled Security Plan reduced the protected area boundary to the Spent Fuel Pit Building outside wall. The area that was within the original protected area boundary (with the exception of the spent fuel complex) was reclassified as an industrial security area. An administrative plant procedure identifies individuals responsible for the management of the security program and for the supervision of the security force.

The NRC has inspected the implementation of the Defueled Security Plan to ensure that the changes had been implemented satisfactorily (Reference 512-4). The inspection concluded that the plan, "as implemented, was directed toward the protection of public health and safety."

To complete the plant decommissioning process, the spent nuclear fuel must be removed from the Spent Fuel Pit. YAEC has constructed an Independent Spent Fuel Storage Installation (ISFSI) onsite and will transfer the spent fuel to this location for interim storage until the Department of Energy is ready to accept this fuel. Since YAEC will store spent fuel under the general license provisions of 10CFR Part 72, Subpart K, YNPS developed an ISFSI Security Plan based on the requirements of 10CFR73.55 with exemptions as appropriate. The YNPS Security Plan amendment submittal (Reference 512-5) requested NRC approval of the ISFSI Security Plan for YNPS.

512.2 Site Access Control

YNPS access control requirements are presented in an administrative plant procedure. This procedure presents requirements for general site access to and access to the protected area(s).

YNPS will maintain a Fitness For Duty program for personnel granted unescorted access to either the general site area or the protected area(s). The objective of the program is to provide a drug and

alcohol free work environment. The program is essential to maintaining the health and safety of those working on site

REFERENCES

512-1 BYR 92-077, Defueled Security and Training and Qualification Plans, J. K. Thayer to M. B. Fairtile (USNRC), August 11, 1992.

512-2 BYR 92-102, Defueled Security and Training and Qualification Plans, J. K. Thayer to M. B. Fairtile (USNRC), October 22, 1992.

512-3 NYR 92-194, Exemptions From Certain Requirements of 10CFR73.55 For The Yankee Nuclear Power Station (YNPS) (TAC No. M84267), M. B. Fairtile to J. M. Grant, November 24, 1992.

512-4 NYR 93-027, NRC Inspection No. 50-29/93-03, J. H. Joyner (USNRC) to J. K. Thayer, March 26, 1993.

512-5 BYR 2000-068, Proposed Amendement to YNPS Security Plan, B. Wood (YAEC) to NRC, dated October 12, 2000.

513 FIRE PROTECTION

513.1 Program Description

On August 20, 1992, the NRC approved a Technical Specification change to remove the fire protection Technical Specifications (Reference 513-1). The fire protection Technical Specifications were replaced with a set of administrative controls. The Fire Protection Technical Requirements Manual (Reference 513-2) was developed to set forth the operational and surveillance requirements of the Fire Protection Plan as approved by NRC Safety Evaluation Reports dated March 15, 1979, and as supplemented October 1, 1980 and August 27, 1986.

The Fire Protection Plan is based on defense in depth. The plan will maintain the following features, as appropriate, during decommissioning:

- Fire detection equipment and systems.
- Personnel training and qualification program.
- Fire Protection Program procedures.
- Control of transient combustible materials and ignition sources, including limitations on inflammable gases as described in Section 406.

Decommissioning activities will be reviewed during the safe storage and dismantlement periods to ensure that the appropriate level of fire protection is being implemented. In addition, fire protection requirements will be evaluated during detailed planning of decommissioning activities.

Administrative plant procedures implement the Fire Protection Plan:

- Defines the Fire Protection Plan for the YNPS, describes the organization and structure of the fire protection program, and defines the administrative and functional responsibilities of assigned personnel.
- Establishes the administrative controls for materials and events that could create potential fire hazards. Materials and processes, which represent a potential fire hazard, are controlled to minimize the possibility of a fire and the effect of a fire on the following:
  - Operation of the SFP cooling and support systems.
  - The release of radioactive material to the environment.
  - Plant personnel safety.
  - Balance of the facility.
- Establishes housekeeping requirements, provides guidelines for hot work, designates fire areas and defines the process for control of combustibles.

513.2 Fire Protection Program Implementation

Continued operability of required fire detection systems ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. The operability of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in those protected facilities. Off-site assistance is provided by the Town of Rowe. The Rowe Fire Department is available within a 30-minute response time.

Fire detection and suppression capabilities will remain in place during the decontamination and dismantlement of contaminated systems and equipment and until fire loading has been substantially reduced. In addition, spatial separation, and the grouping by activity level will be used to minimize the radiological consequences of a fire in radioactive materials awaiting shipment from the site. Prior to final building and area decontamination and dismantlement activities, fire detection and suppression systems will be isolated and removed. Supplemental fire detection and suppression measures may be employed as required.

REFERENCES

513-1 NYR 92-156, Issuance of Amendment No. 142 To Facility Possession Only License No. DPR-3 - Yankee Nuclear Power Station (TAC No. M83746), M. B. Fairtile (USNRC) to J. M. Grant, August 20, 1992.

513-2 Fire Protection Technical Requirements Manual.

514 DECOMMISSIONING QUALITY ASSURANCE PROGRAM

514.1 Description

Yankee Atomic Electric Company (YAEC) has developed and implemented a comprehensive Quality Assurance Program to assure conformance with established regulatory requirements set forth by the Nuclear Regulatory Commission (NRC) and accepted industry standards. The participants in the Yankee Decommissioning Quality Assurance Program (YDQAP) assure that the storage of spent fuel and the decommissioning of the Yankee Nuclear Power Station are performed in a safe and effective manner.

The YDQAP complies with the requirements set forth in Appendix B of 10CFR Part 50, along with applicable sections of the Updated Final Safety Analysis Report (UFSAR) for the license application, and is responsive to Regulatory Guide 1.70.

The YDQAP is also established, maintained and executed to comply with the requirements of 10CFR71, Subpart H, and 10CFR72, Subpart G for the storage and transportation of spent nuclear fuel and high level waste under the provisions of a General License contained in these parts.

The YDQAP is submitted periodically to the NRC in accordance with 10CFR50.54(a).

REFERENCES

514-1 License No. DPR-3 - Yankee Nuclear Power Station

514-2 YDQAP; Yankee Decommissioning Quality Assurance Program

514-3 10CFR Part 50 Appendix B; Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

514-4 NRC Inspection and Enforcement Manual, Chapter 2561; Reactor Inspection Program - Post-Operational Phase

515 EMERGENCY PLAN

The Defueled Emergency Plan for YNPS was approved by the NRC and issued on October 30, 1992. The Defueled Emergency Plan is implemented through the use of Emergency Implementing Procedures which are referenced in Appendix B of the Plan. The Defueled Emergency Plan is formally audited as part of the Quality Assurance Program.

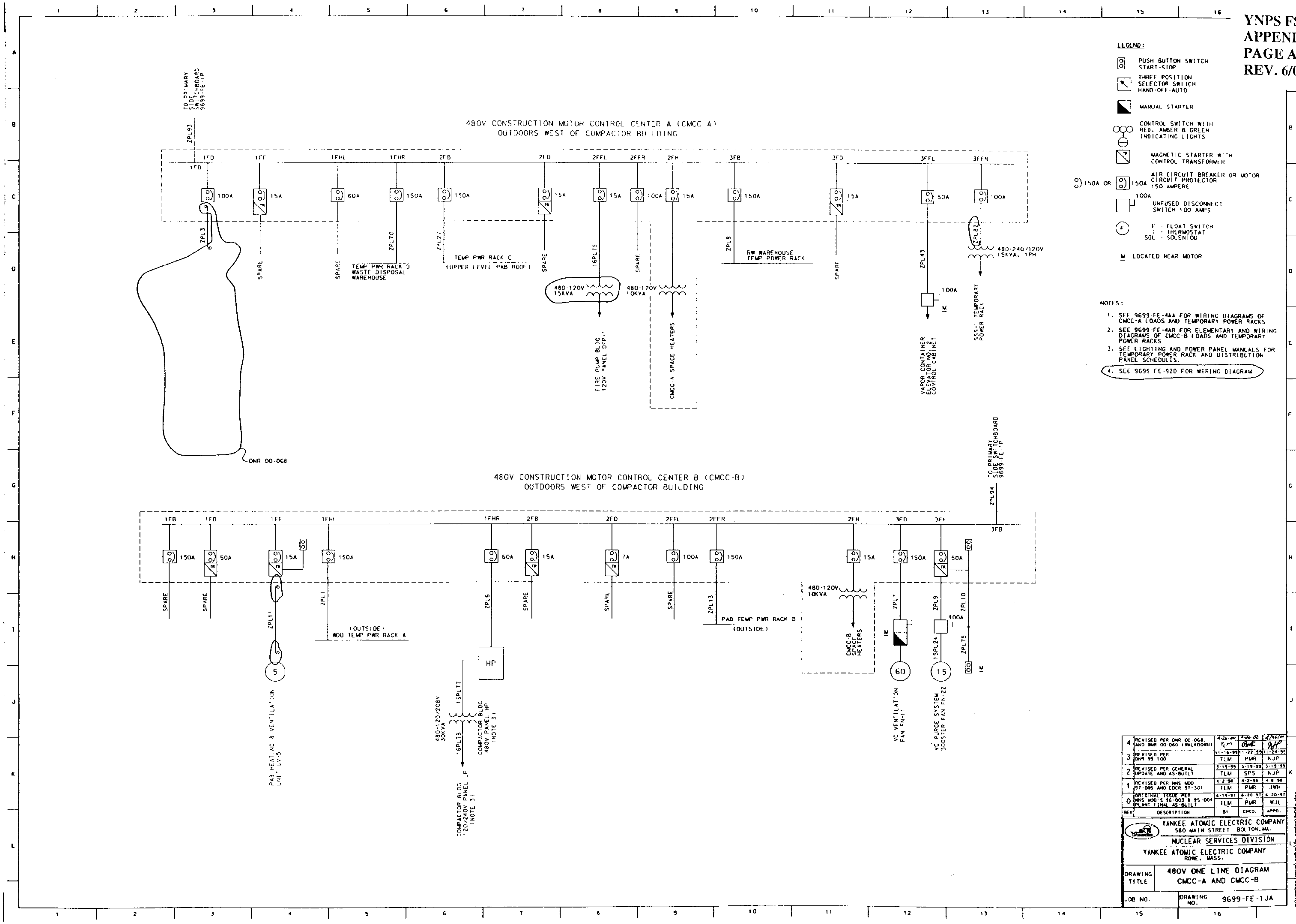
APPENDIX A

Plant Drawings

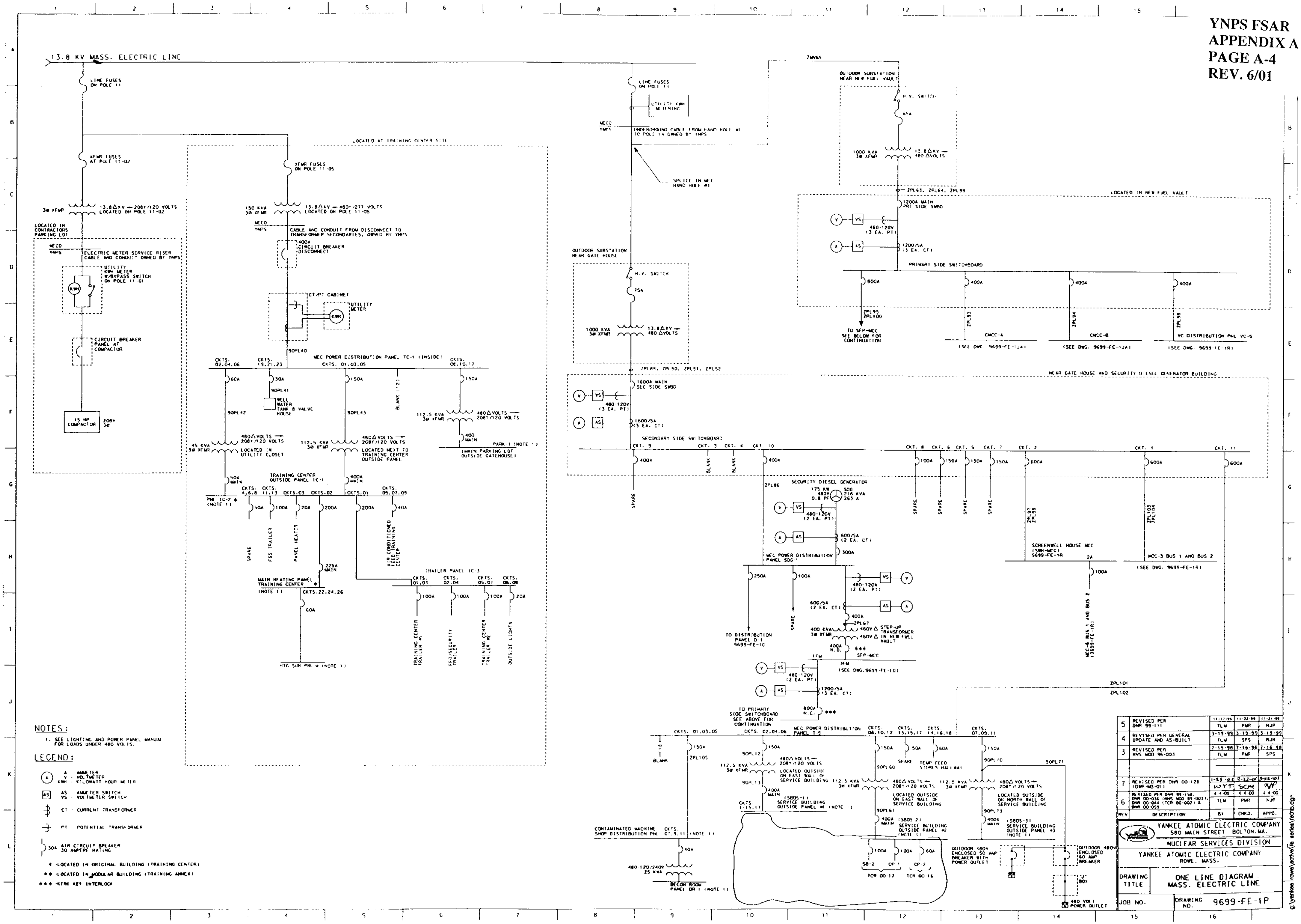
APPENDIX A

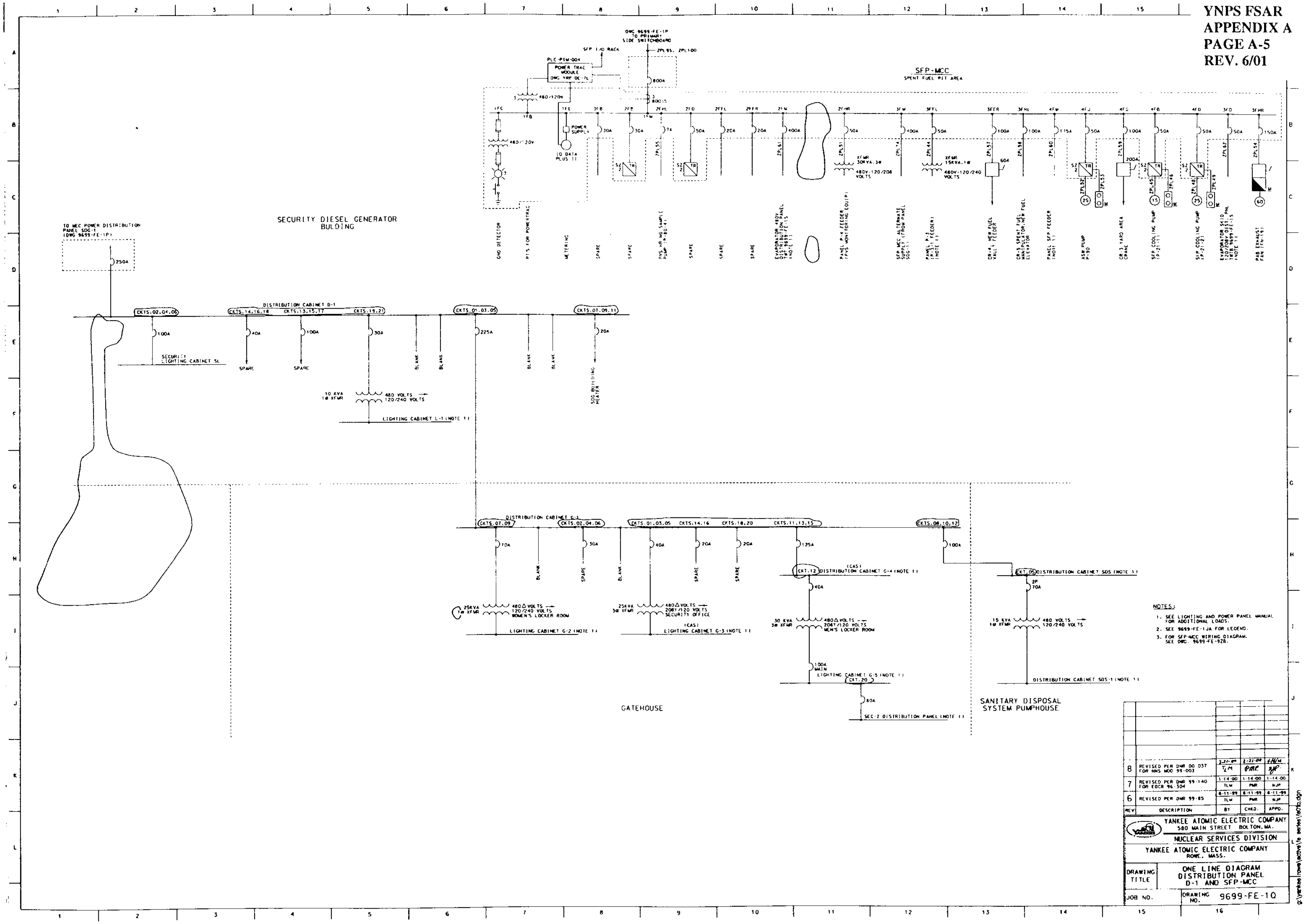
Plant Drawings

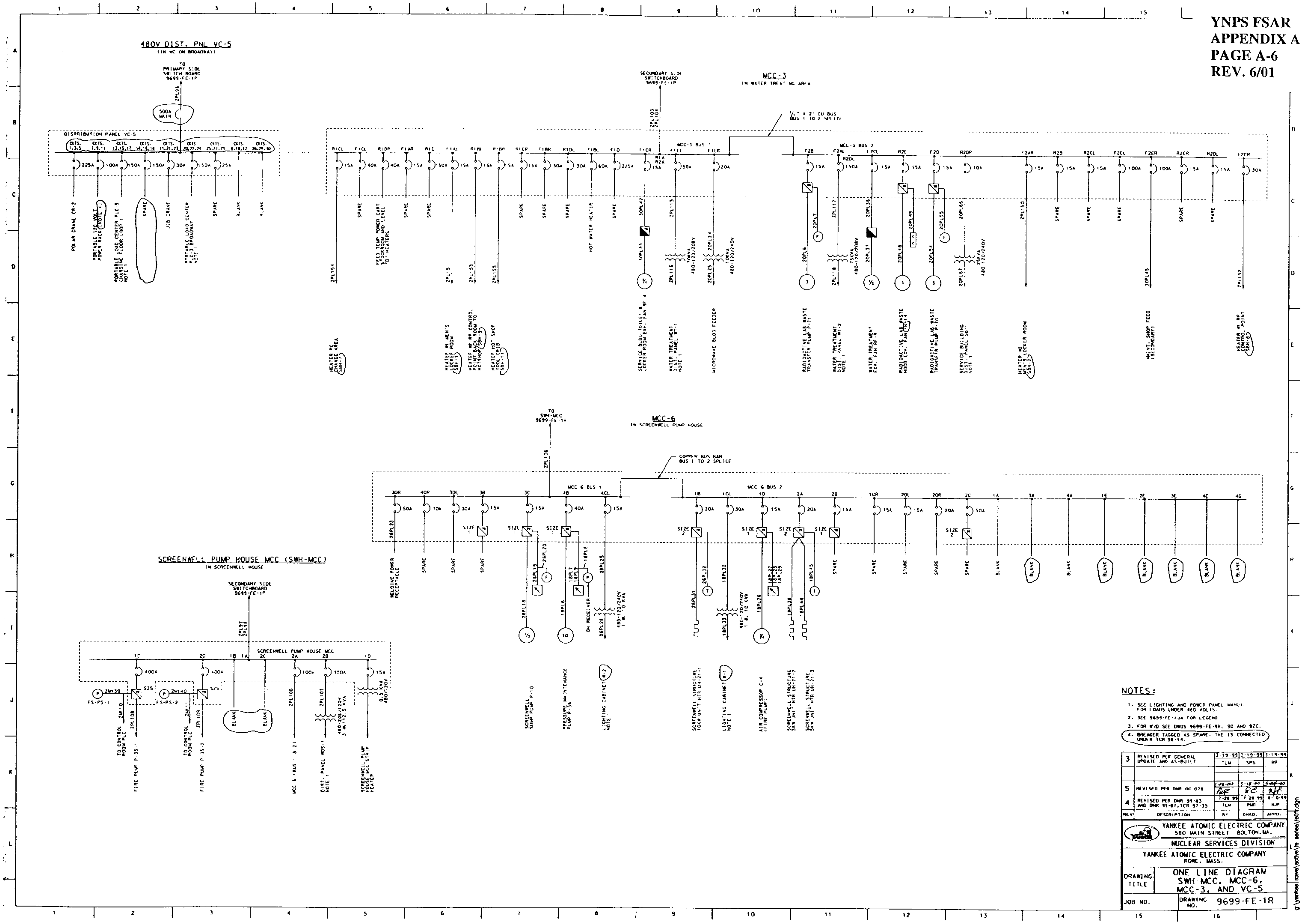
*Note date of last revision; subsequent revisions may have occurred since printing of FSAR.

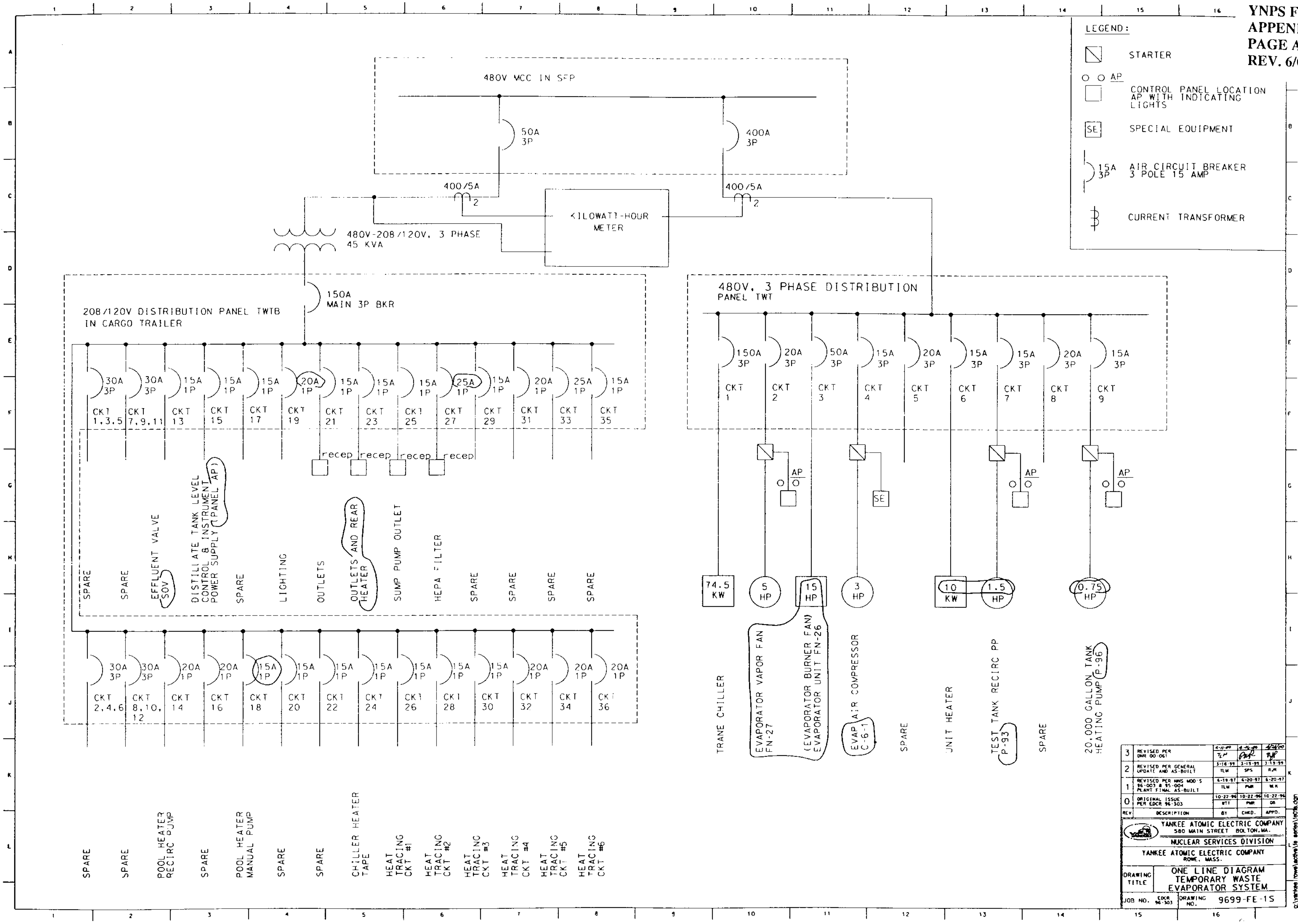

480V CONSTRUCTION MOTOR CONTROL CENTER A (CMCC-A)
OUTDOORS WEST OF COMPACTOR BUILDING
TO PRIMARY SIDE SWITCHBOARD 9699-E-1P
ZPL93
1FD
1FF
1FHL
1FHR
2FB
2FD
2FFL
2FFR
2FH
3FB
3FD
3FFL
3FFR
1FB
100A
15A
60A
150A
50A
SPARE
TEMP PWR RACK D WASTE DISPOSAL WAREHOUSE
TEMP PWR RACK C (UPPER LEVEL PAB ROOF)
RW WAREHOUSE TEMP POWER RACK
480-120V 15KVA
480-120V 10KVA
480-240/120V 15KVA, 1PH
FIRE PUMP BLDG 120V PANEL DFP-1
CMCC-A SPACE HEATERS
VAPOR CONTAINER ELEVATOR NO. 2 CONTROL CABINET
SSS-1 TEMPORARY POWER RACK
DNR 00-068
480V CONSTRUCTION MOTOR CONTROL CENTER B (CMCC-B)
OUTDOORS WEST OF COMPACTOR BUILDING
TO PRIMARY SIDE SWITCHBOARD 9699-FE-1P
ZPL94
1FB
1FD
1FF
1FHL
1FHR
2FB
2FD
2FFL
2FFR
2FH
3FD
3FF
3FB
150A
50A
15A
60A
7A
100A
SPARE
(OUTSIDE) WOB TEMP PWR RACK A
PAB TEMP PWR RACK B (OUTSIDE)
480-120V 10KVA
CMCC-B SPACE HEATERS
HP
480-120/208V 30KVA
PAB HEATING & VENTILATION UNIT UV-5
COMPACTOR BLDG 480V PANEL HP (NOTE 3)
COMPACTOR BLDG 120/240V PANEL LP (NOTE 3)
VC VENTILATION FAN FN-11
VC PURGE SYSTEM BOOSTER FAN FN-22
LEGEND:
PUSH BUTTON SWITCH START-STOP
THREE POSITION SELECTOR SWITCH HAND-OFF-AUTO
MANUAL STARTER
CONTROL SWITCH WITH RED, AMBER & GREEN INDICATING LIGHTS
MAGNETIC STARTER WITH CONTROL TRANSFORMER
150A OR 150A AIR CIRCUIT BREAKER OR MOTOR CIRCUIT PROTECTOR 150 AMPERE
100A UNFUSED DISCONNECT SWITCH 100 AMPS
F - FLOAT SWITCH
T - THERMOSTAT
SOL - SOLENOID
M LOCATED NEAR MOTOR
NOTES:
1. SEE 9699-FE-4AA FOR WIRING DIAGRAMS OF CMCC-A LOADS AND TEMPORARY POWER RACKS
2. SEE 9699-FE-4AB FOR ELEMENTARY AND WIRING DIAGRAMS OF CMCC-B LOADS AND TEMPORARY POWER RACKS
3. SEE LIGHTING AND POWER PANEL MANUALS FOR TEMPORARY POWER RACK AND DISTRIBUTION PANEL SCHEDULES.
4. SEE 9699-FE-9ZD FOR WIRING DIAGRAM
4 REVISED PER DNR 00-068, AND DNR 00-060 (WALKDOWN)
3 REVISED PER DNR 99-100
2 REVISED PER GENERAL UPDATE AND AS-BUILT
1 REVISED PER MMS MOD 97-005 AND EDCR 97-301
0 ORIGINAL ISSUE PER MMS MOD'S 96-003 & 95-004 PLANT FINAL AS-BUILT
REV. DESCRIPTION BY CHKD. APPD.
YANKEE ATOMIC ELECTRIC COMPANY 580 MAIN STREET BOLTON, MA.
NUCLEAR SERVICES DIVISION
YANKEE ATOMIC ELECTRIC COMPANY ROWE, MASS.
DRAWING TITLE 480V ONE LINE DIAGRAM CMCC-A AND CMCC-B
JOB NO.
DRAWING NO. 9699-FE-1JA

YNPS FSAR
APPENDIX A
PAGE A-4
REV. 6/01

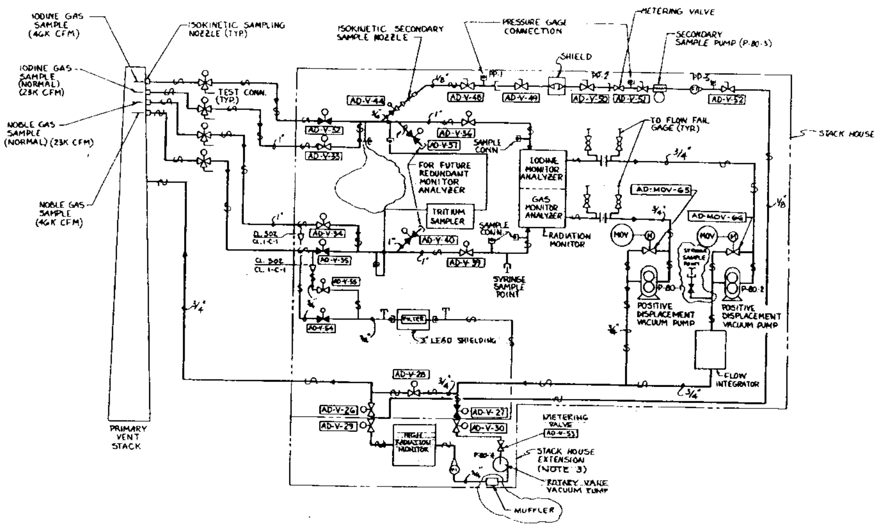



SFP-MCC
SPENT FUEL PIT AREA
SECURITY DIESEL GENERATOR BULDING
DISTRIBUTION CABINET D-1
SECURITY LIGHTING CABINET SL
SPARE
BLANK
10 KVA 1Ø XFMR
480 VOLTS 120/240 VOLTS
LIGHTING CABINET L-1 (NOTE 1)
SDG BUILDING HEATER
DISTRIBUTION CABINET G-1
LIGHTING CABINET G-2 (NOTE 1)
LIGHTING CABINET G-3 (NOTE 1)
DISTRIBUTION CABINET G-4 (NOTE 1)
LIGHTING CABINET G-5 (NOTE 1)
SEC-2 DISTRIBUTION PANEL (NOTE 1)
DISTRIBUTION CABINET SDS (NOTE 1)
DISTRIBUTION CABINET SDS-1 (NOTE 1)
GATEHOUSE
SANITARY DISPOSAL SYSTEM PUMPHOUSE
NOTES:
1. SEE LIGHTING AND POWER PANEL MANUAL FOR ADDITIONAL LOADS.
2. SEE 9699-FE-1JA FOR LEGEND.
3. FOR SFP-MCC WIRING DIAGRAM, SEE DWG. 9699-FE-92B.
YANKEE ATOMIC ELECTRIC COMPANY
580 MAIN STREET BOLTON, MA.
NUCLEAR SERVICES DIVISION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
DRAWING TITLE
ONE LINE DIAGRAM
DISTRIBUTION PANEL
D-1 AND SFP-MCC
JOB NO.
DRAWING NO.
9699-FE-1Q


480V DIST. PNL VC-5
DISTRIBUTION PANEL VC-5
MCC-3
IN WATER TREATING AREA
MCC-3 BUS 1
MCC-3 BUS 2
MCC-6
IN SCREENWELL PUMP HOUSE
MCC-6 BUS 1
MCC-6 BUS 2
COPPER BUS BAR BUS 1 TO 2 SPLICE
SCREENWELL PUMP HOUSE MCC (SWH-MCC)
IN SCREENWELL HOUSE
SECONDARY SIDE SWITCHBOARD 9699-FE-1P
NOTES:
1. SEE LIGHTING AND POWER PANEL MANUAL FOR LOADS UNDER 480 VOLTS.
2. SEE 9699-FE-1JA FOR LEGEND
3. FOR W/D SEE DWGS 9699-FE-9H, 9D AND 9ZC.
4. BREAKER TAGGED AS SPARE. THE 15 CONNECTED UNDER TCR 98-14.
YANKEE ATOMIC ELECTRIC COMPANY
580 MAIN STREET BOLTON, MA.
NUCLEAR SERVICES DIVISION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
DRAWING TITLE
ONE LINE DIAGRAM
SWH-MCC, MCC-6, MCC-3, AND VC-5
JOB NO.
DRAWING NO. 9699-FE-1R


LEGEND:
STARTER
AP
CONTROL PANEL LOCATION AP WITH INDICATING LIGHTS
SE SPECIAL EQUIPMENT
15A 3P AIR CIRCUIT BREAKER 3 POLE 15 AMP
CURRENT TRANSFORMER
480V MCC IN SFP
50A 3P
400A 3P
400/5A
KILOWATT-HOUR METER
480V-208/120V, 3 PHASE 45 KVA
150A MAIN 3P BKR
208/120V DISTRIBUTION PANEL TWTB IN CARGO TRAILER
30A 3P CKT 1,3,5 SPARE
30A 3P CKT 7,9,11 SPARE
15A 1P CKT 13 EFFLUENT VALVE SOV
15A 1P CKT 15 DISTILLATE TANK LEVEL CONTROL & INSTRUMENT POWER SUPPLY (PANEL AP)
15A 1P CKT 17 SPARE
20A 1P CKT 19 LIGHTING
15A 1P CKT 21 recep OUTLETS
15A 1P CKT 23 recep OUTLETS AND REAR HEATER
15A 1P CKT 25 recep SUMP PUMP OUTLET
25A 1P CKT 27 recep HEPA FILTER
15A 1P CKT 29 SPARE
20A 1P CKT 31 SPARE
25A 1P CKT 33 SPARE
15A 1P CKT 35 SPARE
30A 3P CKT 2,4,6 SPARE
30A 3P CKT 8,10,12 SPARE
20A 1P CKT 14 POOL HEATER RECIRC PUMP
20A 1P CKT 16 SPARE
15A 1P CKT 18 POOL HEATER MANUAL PUMP
15A 1P CKT 20 SPARE
15A 1P CKT 22 SPARE
15A 1P CKT 24 CHILLER HEATER TAPE
15A 1P CKT 26 HEAT TRACING CKT #1
15A 1P CKT 28 HEAT TRACING CKT #2
15A 1P CKT 30 HEAT TRACING CKT #3
20A 1P CKT 32 HEAT TRACING CKT #4
20A 1P CKT 34 HEAT TRACING CKT #5
20A 1P CKT 36 HEAT TRACING CKT #6
480V, 3 PHASE DISTRIBUTION PANEL TWT
150A 3P CKT 1 74.5 KW TRANE CHILLER
20A 3P CKT 2 5 HP EVAPORATOR VAPOR FAN FN-27
50A 3P CKT 3 15 HP (EVAPORATOR BURNER FAN) EVAPORATOR UNIT FN-26
15A 3P CKT 4 SE 3 HP AIR COMPRESSOR EVAP C-6-1
20A 3P CKT 5 SPARE
15A 3P CKT 6 10 KW UNIT HEATER
15A 3P CKT 7 1.5 HP TEST TANK RECIRC PP P-93
20A 3P CKT 8 SPARE
15A 3P CKT 9 0.75 HP 20,000 GALLON TANK HEATING PUMP P-96
3 REVISED PER DMR 00-061
2 REVISED PER GENERAL UPDATE AND AS-BUILT
1 REVISED PER NMS MOD'S 96-003 & 95-004 PLANT FINAL AS-BUILT
0 ORIGINAL ISSUE PER EDCR 96-303
REV DESCRIPTION BY CHKD. APPD.
YANKEE ATOMIC ELECTRIC COMPANY 580 MAIN STREET BOLTON, MA.
NUCLEAR SERVICES DIVISION
YANKEE ATOMIC ELECTRIC COMPANY ROWE, MASS.
DRAWING TITLE ONE LINE DIAGRAM TEMPORARY WASTE EVAPORATOR SYSTEM
JOB NO. EDCR 96-303 DRAWING NO. 9699-FE-1S

FK-13A



NOTES:

1. ALL LINES SHOWN AS 1" ARE CLASS 302 PIPING AND FITTINGS AS SPECIFIED PER YANKEE PIPE SPEC YS-497
2. ALL LINES SHOWN AS 3/4" OR SMALLER ARE CLASS 1-C-1 STAINLESS STEEL TUBING AND FITTINGS AS SPECIFIED PER YANKEE PIPE SPEC YS-497
3. EQUIPMENT LOCATED IN STACK HOUSE ADDITION (HIGH RAD. MONITOR), EXISTING BUT NOT OPERATING.
4. FOUR ISOKINETIC SAMPLING VALVES AT PRIMARY VENT STACK NOT VERIFIED, INACCESSIBLE 5-16-00

REFERENCE DWGS.

STACK MOD. SYS. PIPING MODS. - 9699-FK-13B & C
ELEC HT. TRACING STACK MON. - 9699-FK-13D

LEGEND:

- 3 WAY BALL VALVE W/TEST CONN. (T.C.)
- BALL VALVE
- QUICK CONNECT
- MOTOR OPERATED VALVE (MOV)
- HEAT TRACING
- SAMPLE CARTRIDGE HOLDER
- PLUG VALVE
- ROTOMETER
- PP PRESSURE POINT

REV.	DESCRIPTION	PREP.	CHKD.	APP'D	QUAL. REVIEW
7	REVISED PER DNR 00-099				
6	REVISED PER GENERAL UPDATE (DNR # 99-29)				
5	REVISED PER EDCR 87-309 AND G.U.				
4	REVISED PER EDCR 84-326				
3	THIS DWG SUPERSEDES 9699-FK-13A REV. 2 PER EDCR 84-314 & EDCR 84-326				

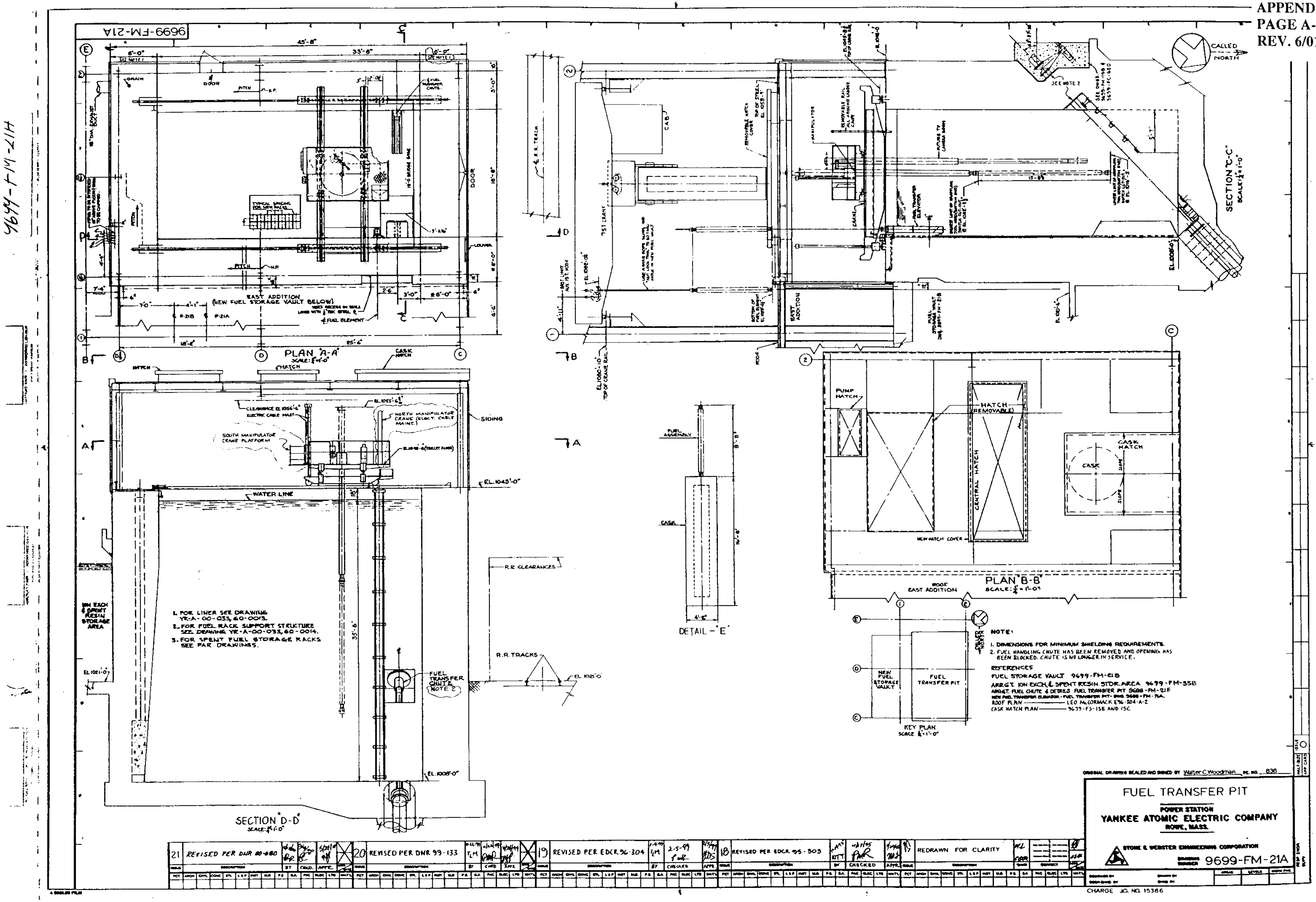
YANKEE ATOMIC ELECTRIC COMPANY

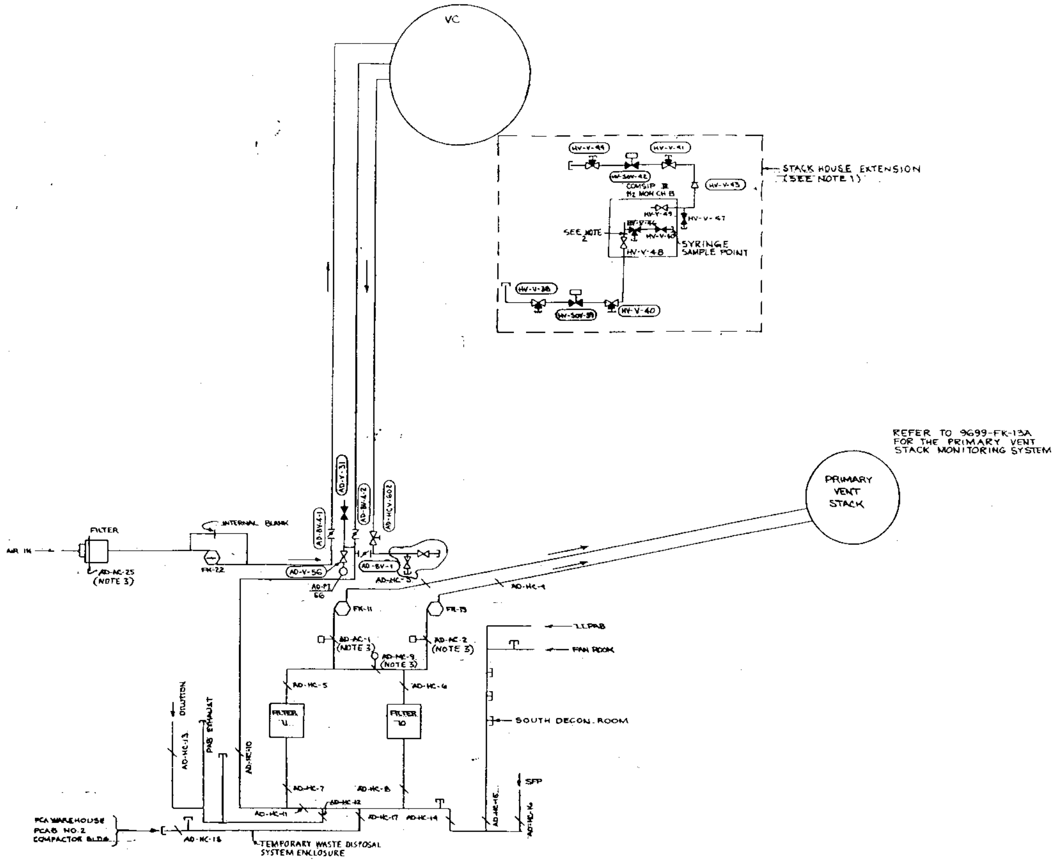
NUCLEAR SERVICES DIVISION

YANKEE ATOMIC ELECTRIC CO.
ROWE, MASS.

P & I DIAGRAM
STACK MONITORING SYSTEM

9699-FK-13A


PLAN "A-A"
SCALE: 1/8"=1'-0"
EAST ADDITION
(NEW FUEL STORAGE VAULT BELOW)
SECTION "C-C"
SCALE: 1/4"=1'-0"
SECTION "D-D"
SCALE: 1/8"=1'-0"
DETAIL - "E"
PLAN "B-B"
SCALE: 1/8"=1'-0"
KEY PLAN
NEW FUEL STORAGE VAULT
FUEL TRANSFER PIT
PUMP HATCH
HATCH (REMOVABLE)
CENTRAL HATCH
CASK HATCH
CASK
NEW HATCH COVER
ROOF EAST ADDITION
WATER LINE
EL. 1045'-0"
EL. 1021'-0"
EL. 1008'-0"
R.R. TRACKS
R.R. CLEARANCES
FUEL TRANSFER CHUTE (NOTE 2)
SOUTH MANIPULATOR CRANE PLATFORM
NORTH MANIPULATOR CRANE (ELECT. CABLE MAINT.)
SIDING
1. FOR LINER SEE DRAWING YR-A-00-033, 60-0013.
2. FOR FUEL RACK SUPPORT STRUCTURE SEE DRAWING YR-A-00-033, 60-0014.
3. FOR SPENT FUEL STORAGE RACKS SEE PAR DRAWINGS.
NOTE:
1. DIMENSIONS FOR MINIMUM SHIELDING REQUIREMENTS.
2. FUEL HANDLING CHUTE HAS BEEN REMOVED AND OPENING HAS BEEN BLOCKED. CHUTE IS NO LONGER IN SERVICE.
REFERENCES
FUEL STORAGE VAULT 9699-FM-21B
ARRGT. ION EXCH. & SPENT RESIN STOR. AREA 9699-FM-55B
ARRGT. FUEL CHUTE & DETAILS FUEL TRANSFER PIT 9699-FM-21F
ROOF PLAN LEO McCORMACK E56-304-A-2
CASK HATCH PLAN 9699-FS-15B AND 15C
21 REVISED PER DNR 00-080
20 REVISED PER DNR 99-133
19 REVISED PER EDCR 96-304
18 REVISED PER EDCR 95-305
17 REDRAWN FOR CLARITY
FUEL TRANSFER PIT
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
9699-FM-21A
CHARGE JO. NO. 15386



PRIMARY SIDE VENTILATION SYSTEM

NOTES:

1. EQUIPMENT IN STACK HOUSE EXTENSION EXISTING BUT NOT OPERATIONAL.
2. HV-V-46 IS A SPECIAL PORTED WHITEY VALVE MODEL SS-43XHF4. ALL PORTS ARE EITHER OPEN OR CLOSED.
3. DAMPERS ARE MANUALLY CONTROLLED (REFER TO EDCR 97-301).

REFERENCE DRAWINGS:

9699-FK-13A, B, C, & D
YR-E-10-001
9699-FE-22H

REV.	DESCRIPTION	PREP.	CHKD.	APP'D	QUAD. REVIEW
16	REVISED PER DWP-WG-01	TLM	WTT	KE	
15	REV. PER C.U. 91-177		BF		
14	REV. PER EDCR 87-309		PC		
20	REVISED PER DNR 00-084	PC			
19	REVISED PER DNR 99-141	TLM			

REV.	DESCRIPTION	BY	CHKD	APP'D
18	REVISED PER G.U. (DNR# 99-58)			
17	[illegible]		TLM	

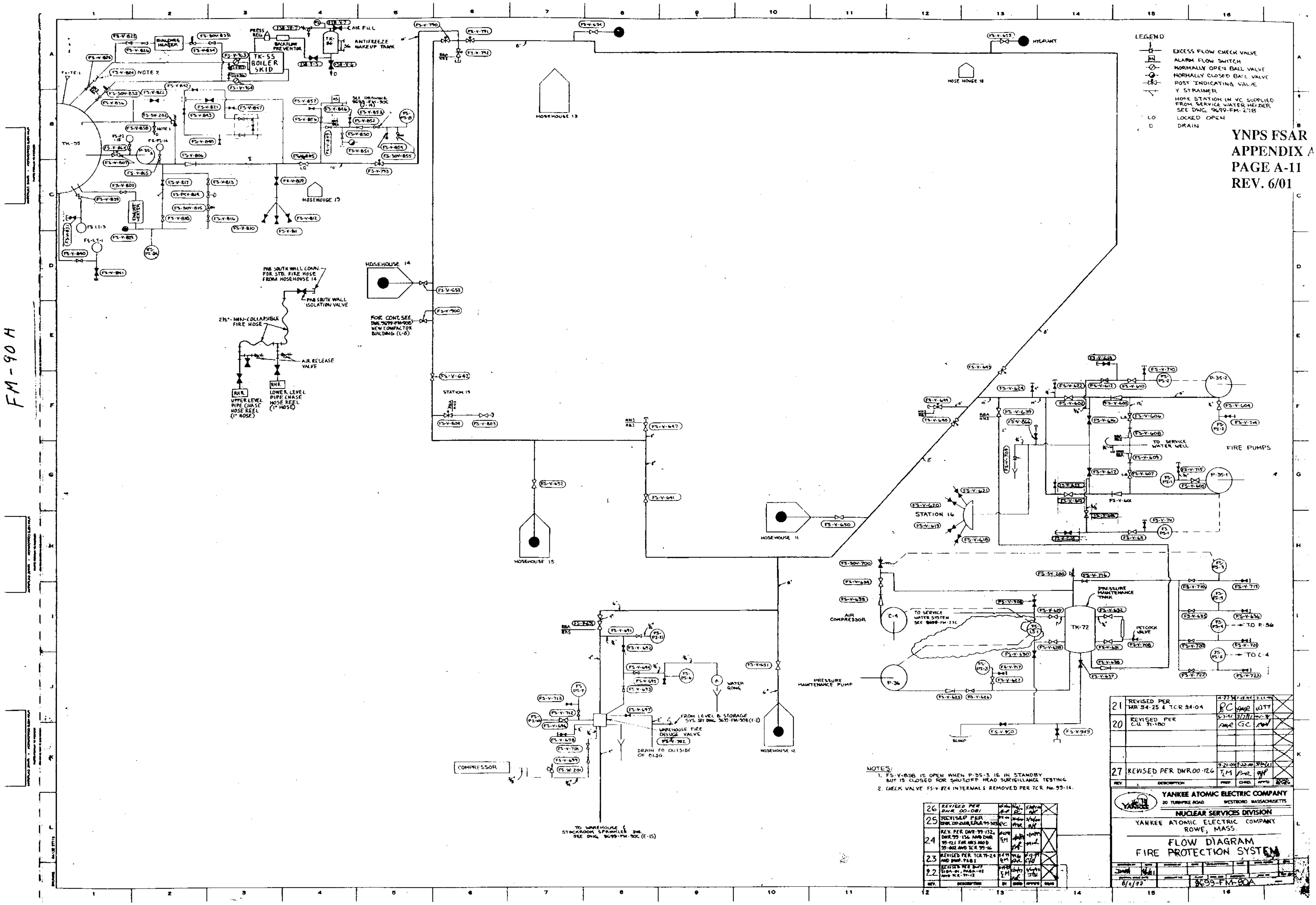
YANKEE ATOMIC ELECTRIC COMPANY
20 TURNPIKE ROAD WESTBORO, MASSACHUSETTS
NUCLEAR SERVICES DIVISION

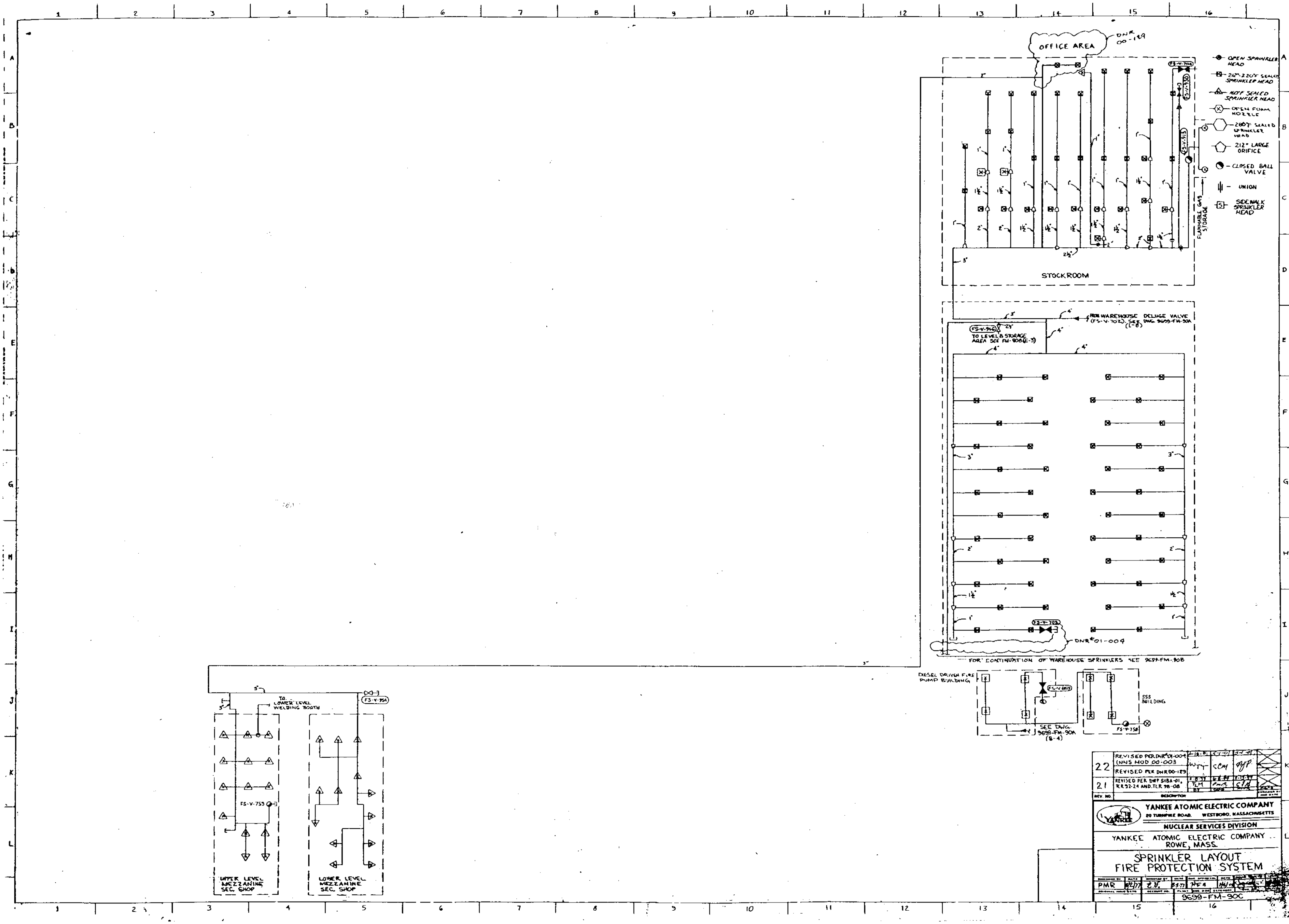
YANKEE ATOMIC ELECTRIC COMPANY
ROWE MA

PRIMARY AIR DISPOSAL PVS

9699 — FM — 26D

FM-26D


FM-90H
LEGEND
EXCESS FLOW CHECK VALVE
ALARM FLOW SWITCH
NORMALLY OPEN BALL VALVE
NORMALLY CLOSED BALL VALVE
POST INDICATING VALVE
Y STRAINER
HOSE STATION IN VC SUPPLIED FROM SERVICE WATER HEADER SEE DWG. 9699-FM-27B
LO LOCKED OPEN
D DRAIN
TK-55 BOILER SKID
ANTIFREEZE MAKEUP TANK
BACKFLOW PREVENTOR
AIR FILL
HOSEHOUSE 13
HOSEHOUSE 14
HOSEHOUSE 15
HOSEHOUSE 11
HOSEHOUSE 12
HOSE HOUSE 18
HYDRANT
STATION 13
STATION 16
FIRE PUMPS
TO SERVICE WATER WELL
PRESSURE MAINTENANCE TANK
TK-72
PETCOCK VALVE
AIR COMPRESSOR
PRESSURE MAINTENANCE PUMP
TO SERVICE WATER SYSTEM SEE 9699-FM-27C
TO P-36
TO C-4
SUMP
PAB SOUTH WALL CONN. FOR STD. FIRE HOSE FROM HOSEHOUSE 14
PAB SOUTH WALL ISOLATION VALVE
2½" NON-COLLAPSIBLE FIRE HOSE
AIR RELEASE VALVE
UPPER LEVEL PIPE CHASE HOSE REEL (1" HOSE)
LOWER LEVEL PIPE CHASE HOSE REEL (1" HOSE)
FOR CONT. SEE DWG. 9699-FM-90B NEW COMPACTOR BUILDING (L-8)
FROM LEVEL B STORAGE SYS. SEE DWG. 9699-FM-90B (I-2)
WAREHOUSE FIRE DELUGE VALVE
DRAIN TO OUTSIDE OF BLDG
WATER GONG
COMPRESSOR
TO WAREHOUSE & STOCKROOM SPRINKLER SYS. SEE DWG. 9699-FM-90C (E-15)
NOTES:
1. FS-V-858 IS OPEN WHEN P-35-3 IS IN STANDBY BUT IS CLOSED FOR SHUTOFF HEAD SURVEILLANCE TESTING
2. CHECK VALVE FS-V-824 INTERNALS REMOVED PER TCR No. 99-14.
21 REVISED PER MR 94-25 & TCR 94-04
20 REVISED PER C.U. 91-180
27 REVISED PER DWR 00-126
REV. DESCRIPTION PREP. CHKD. APPD.
26 REVISED PER DWR 00-081
25 REVISED PER DWR 00-... EPCR 99-303
24 REV. PER DWR-99-132, DWR 99-136 AND DWR 99-121 FOR NHS MOD 99-602 AND TCR 99-16
23 REVISED PER TCR 97-24 AND DWP-PAB 1
22 REVISED PER DWP SIBA-01, PABA-03 AND TCR-97-13
REV. DESCRIPTION BY CHKD APPD
YANKEE ATOMIC ELECTRIC COMPANY
20 TURNPIKE ROAD WESTBORO, MASSACHUSETTS
NUCLEAR SERVICES DIVISION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
FLOW DIAGRAM
FIRE PROTECTION SYSTEM
8/2/77
9699-FM-90A



REV. NO.	DESCRIPTION			
22	REVISED PER DNR 01-004 (NNS MOD 00-003; REVISED PER DNR 00-129	WTT	SCM	YJP
21	REVISED PER DWP SIBA-01, RR 92-24 AND TLR 98-08	TLM	PMR	CTK

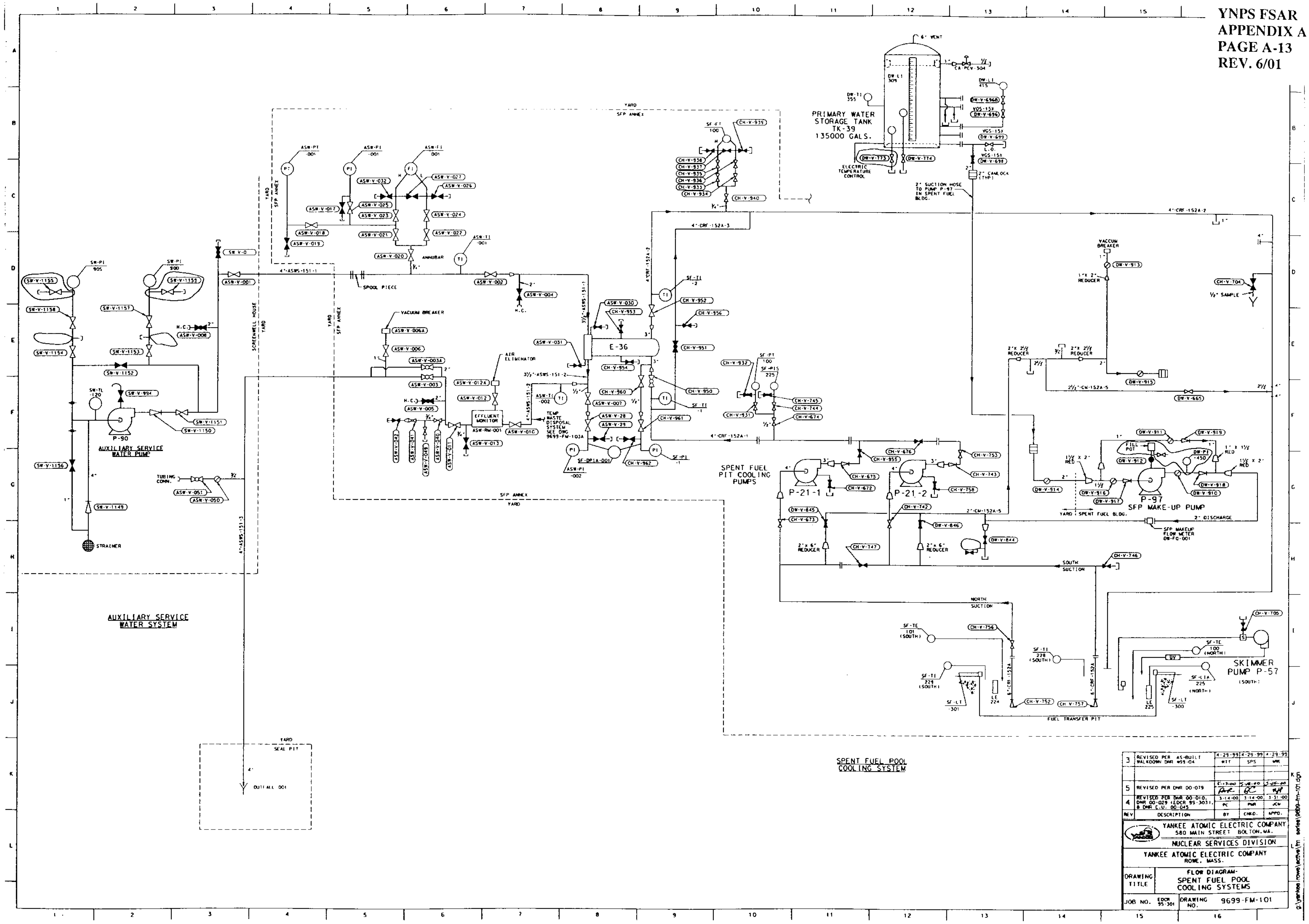
YANKEE ATOMIC ELECTRIC COMPANY
20 TURNPIKE ROAD. WESTBORO, MASSACHUSETTS
NUCLEAR SERVICES DIVISION

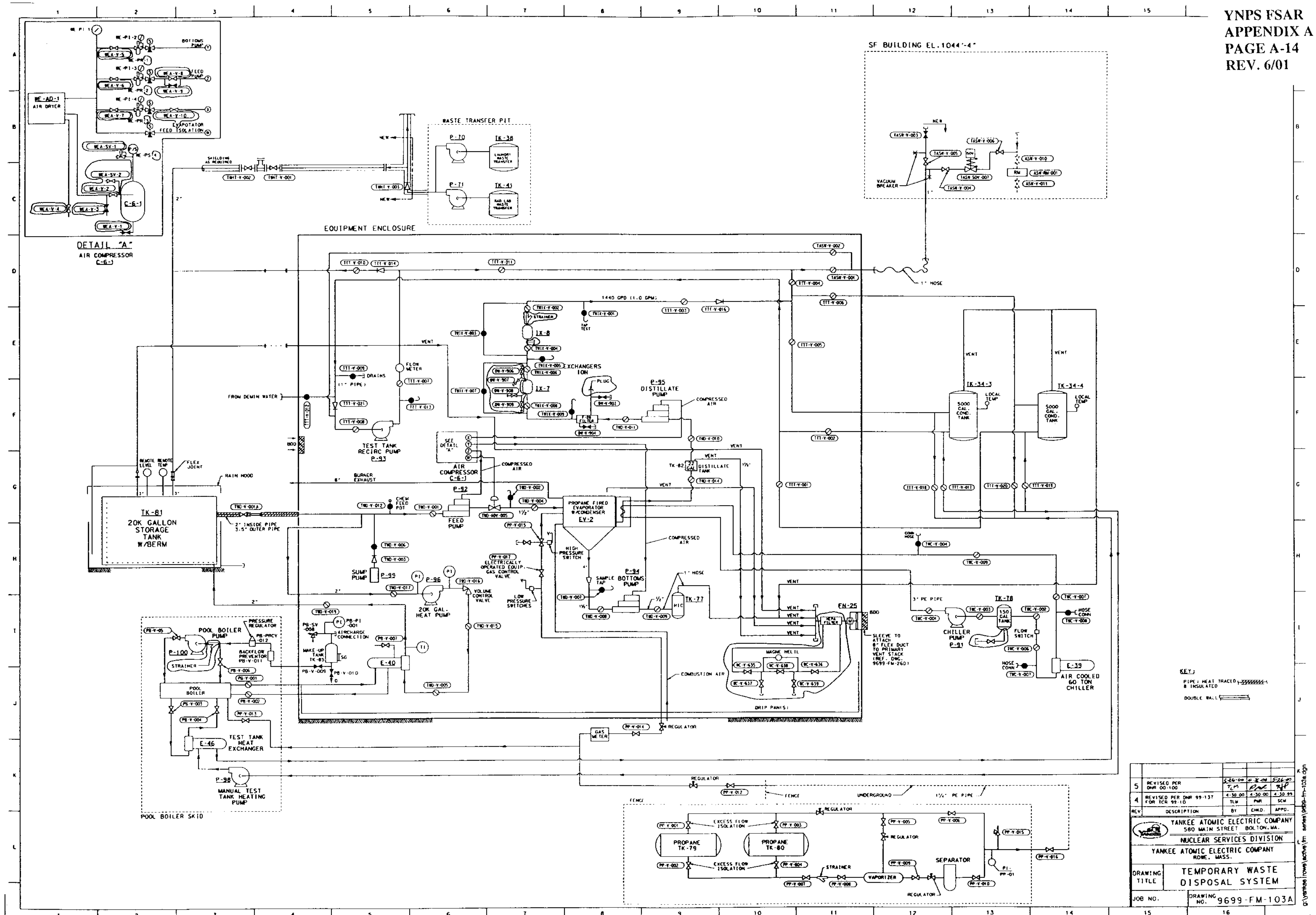
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.

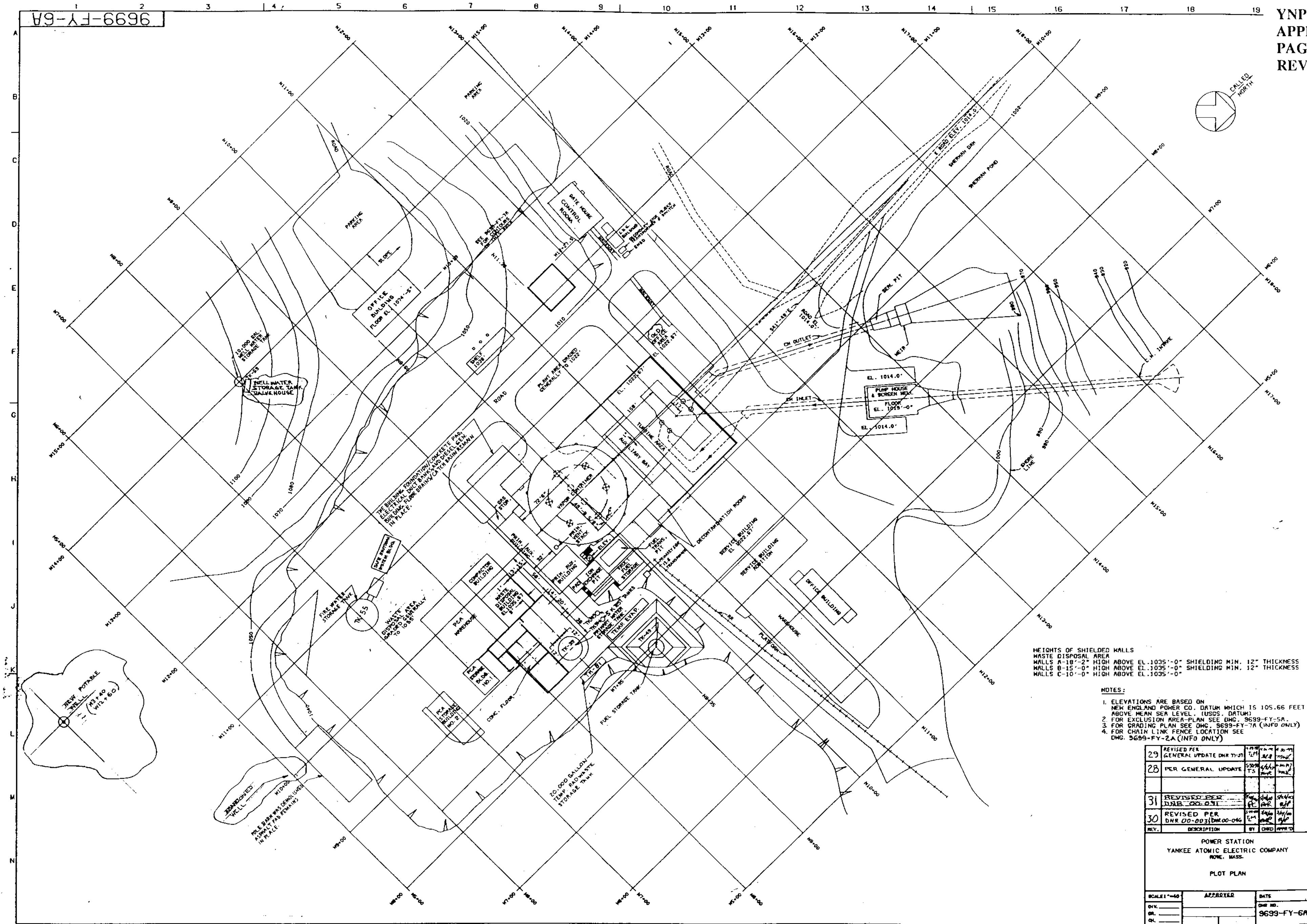
SPRINKLER LAYOUT
FIRE PROTECTION SYSTEM

PMR 4/2/77 E.V. 5-5-77

9699-FM-90C


PRIMARY WATER STORAGE TANK TK-39 135000 GALS.
AUXILIARY SERVICE WATER PUMP
P-90
STRAINER
AUXILIARY SERVICE WATER SYSTEM
SPOOL PIECE
VACUUM BREAKER
AIR ELIMINATOR
EFFLUENT MONITOR
ANNUBAR
TEMP WASTE DISPOSAL SYSTEM SEE DWG 9699-FM-103A
E-36
SPENT FUEL PIT COOLING PUMPS
P-21-1
P-21-2
SFP MAKE-UP PUMP
P-97
FILL POT
SOUTH SUCTION
NORTH SUCTION
FUEL TRANSFER PIT
SKIMMER PUMP P-57
SPENT FUEL POOL COOLING SYSTEM
SEAL PIT
OUTFALL 001
YARD
SFP ANNEX
SCREENWELL HOUSE
3 REVISED PER AS-BUILT WALKDOWN DWR #95-04
5 REVISED PER DWR 00-079
4 REVISED PER DWR 00-010, DWR 00-029 (EDCR 95-303), & DWR C.U. 00-045
REV DESCRIPTION BY CHKD. APPD.
YANKEE ATOMIC ELECTRIC COMPANY 580 MAIN STREET BOLTON, MA.
NUCLEAR SERVICES DIVISION
YANKEE ATOMIC ELECTRIC COMPANY ROWE, MASS.
DRAWING TITLE
FLOW DIAGRAM- SPENT FUEL POOL COOLING SYSTEMS
JOB NO. EDCR 95-301
DRAWING NO. 9699-FM-101


SF BUILDING EL.1044'-4"
WASTE TRANSFER PIT
P-70
TK-38
LAUNDRY WASTE TRANSFER
P-71
TK-43
RAD LAB WASTE TRANSFER
DETAIL "A"
AIR COMPRESSOR
C-6-1
EQUIPMENT ENCLOSURE
TK-81
20K GALLON STORAGE TANK W/BERM
2" INSIDE PIPE
3.5" OUTER PIPE
RAIN HOOD
FLEX JOINT
REMOTE LEVEL
REMOTE TEMP
FROM DEMIN WATER
TEST TANK RECIRC PUMP
P-93
FLOW METER
DRAINS
IX-8
IX-7
ION EXCHANGERS
P-95
DISTILLATE PUMP
COMPRESSED AIR
1440 GPD (1.0 GPM)
AIR COMPRESSOR
C-6-1
SEE DETAIL "A"
BURNER EXHAUST
CHEM FEED POT
P-92
FEED PUMP
PROPANE FIRED EVAPORATOR W/CONDENSER
EV-2
HIGH PRESSURE SWITCH
TK-82
22 GAL DISTILLATE TANK
SUMP PUMP
P-99
P-96
20K GAL. HEAT PUMP
VOLUME CONTROL VALVE
ELECTRICALLY OPERATED EQUIP. GAS CONTROL VALVE
LOW PRESSURE SWITCHES
SAMPLE TAP
P-94
BOTTOMS PUMP
TK-77
FN-25
HEPA FILTER
SLEEVE TO ATTACH 8" FLEX DUCT TO PRIMARY VENT STACK (REF. DWG. 9699-FM-260)
DRIP PANS
COMBUSTION AIR
TK-34-3
TK-34-4
5000 GAL. COND. TANK
LOCAL TEMP
1" HOSE
3" PE PIPE
CHILLER PUMP
P-91
TK-78
150 GAL TANK
FLOW SWITCH
HOSE CONN
E-39
AIR COOLED 60 TON CHILLER
POOL BOILER PUMP
P-100
PRESSURE REGULATOR
BACKFLOW PREVENTOR
STRAINER
POOL BOILER
MAKE-UP TANK TK-85
AIRCHARGE CONNECTION
E-40
E-46
TEST TANK HEAT EXCHANGER
P-98
MANUAL TEST TANK HEATING PUMP
POOL BOILER SKID
GAS METER
REGULATOR
FENCE
UNDERGROUND
1¼" PE PIPE
PROPANE TK-79
PROPANE TK-80
EXCESS FLOW ISOLATION
VAPORIZER
SEPARATOR
KEY:
PIPE: HEAT TRACED & INSULATED
DOUBLE WALL
5 REVISED PER DMR 00-100
4 REVISED PER DMR 99-137 FOR TCR 99-10
REV DESCRIPTION BY CHKD. APPD.
YANKEE ATOMIC ELECTRIC COMPANY
580 MAIN STREET BOLTON, MA.
NUCLEAR SERVICES DIVISION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
DRAWING TITLE
TEMPORARY WASTE DISPOSAL SYSTEM
JOB NO.
DRAWING NO. 9699-FM-103A


9699-FY-6A
CALLED NORTH
GATE HOUSE CONTROL ROOM
PARKING AREA
OFFICE BUILDING FLOOR EL. 1074'-5"
10,000 GAL. WELL WATER STORAGE TANK
WELL WATER STORAGE TANK VALVE HOUSE
PLANT AREA GRADED GENERALLY TO 1022'
OLD OFFICE AREA
TURBINE AREA
AUXILIARY BAY
VAPOR CONTAINER
SERVICE BUILDING EL. 1022'-6"
SERVICE BUILDING ADDITION
DECONTAMINATION ROOMS
OFFICE BUILDING
WAREHOUSE
PLATFORM
PCA WAREHOUSE
COMPACTOR BUILDING
FIRE WATER STORAGE TANK
WASTE DISPOSAL AREA GRADED GENERALLY TO 1035'
TEMP. EVAP.
FUEL STORAGE TANK
20,000 GALLON TEMP RAD WASTE STORAGE TANK
CONC. FLOOR
PCA STORAGE BLDG. NO. 1
PCA STORAGE BLDG. NO. 2
NEW POTABLE WELL
ABANDONED WELL
PUMP HOUSE & SCREEN WELL FLOOR EL. 1019'-0"
EL. 1014.0'
CW INLET
CW OUTLET
SEAL PIT
SHERMAN DAM
SHERMAN POND
C.W. INTAKE
SHORE LINE
ROAD
HEIGHTS OF SHIELDED WALLS
WASTE DISPOSAL AREA
WALLS A-18'-2" HIGH ABOVE EL.1035'-0" SHIELDING MIN. 12" THICKNESS
WALLS B-15'-0" HIGH ABOVE EL.1035'-0" SHIELDING MIN. 12" THICKNESS
WALLS C-10'-0" HIGH ABOVE EL.1035'-0"
NOTES:
1. ELEVATIONS ARE BASED ON NEW ENGLAND POWER CO. DATUM WHICH IS 105.66 FEET ABOVE MEAN SEA LEVEL. (USGS. DATUM)
2. FOR EXCLUSION AREA-PLAN SEE DWG. 9699-FY-5A.
3. FOR GRADING PLAN SEE DWG. 9699-FY-7A (INFO ONLY)
4. FOR CHAIN LINK FENCE LOCATION SEE DWG. 9699-FY-2A (INFO ONLY)
29 REVISED PER GENERAL UPDATE DNR 97-37
28 PER GENERAL UPDATE
31 REVISED PER DNR 00-031
30 REVISED PER DNR 00-003 (DNR 00-046
REV. DESCRIPTION BY CHKD APPR'D
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
PLOT PLAN
SCALE 1"=40'
APPROVED
DATE
DWG. NO.
9699-FY-6A