February 20, 1998

Docket No. 50-309

Mr. Michael B. Sellman, President
Maine Yankee Atomic Power Company
Baily Point Road
Wiscasset, Maine 04578

SUBJECT: MEETING SUMMARY - MAINE YANKEE ATOMIC POWER COMPANY, MEETING NO. 98-02, JANUARY 28, 1998

Dear Mr. Sellman:

Representatives from Maine Yankee Atomic Power Company met with NRC Region I staff from the Division of Nuclear Material Safety (DNMS) and NRC headquarters staff from the Office of Nuclear Reactor Regulation on January 28, 1998 in the Region I Public Meeting Room. The purpose of the management meeting was to discuss the licensee's approach to the upcoming decontamination of the reactor system, the design for the spent fuel pool island concept, and the licensee's justification for safety system re-classification at the Maine Yankee plant. The meeting, which was open to public observation, provided useful dialogue to enable the staff to better understand licensee planned actions and their basis for the Maine Yankee plant.

The status of licensee submittals and determinations regarding safety classification of systems and equipment at the plant were discussed. The licensee has concluded that much of the plant is no longer considered as "important to safety" because of the reduced risk the plant presents in its current condition. The licensee presented to NRC staff the bases for their conclusions, and introduced a definition for equipment classified as "Important to the Defueled Condition" (ITDC).

The licensee also described conceptual plans and work-in-progress for developing a spent fuel pool island that reduces or eliminates the reliance on a number of large plant systems currently used for spent fuel pool cooling. The intent presented was to minimize the physical size of the area required to maintain the spent fuel pool, and to modify or replace any required existing plant equipment as necessary to support the new configuration. Engineering bases and plans for the project were discussed.

The licensee is planning to perform a chemical decontamination of the reactor coolant system to remove as much radioactivity as possible from the systems prior to dismantlement, in order to reduce associated personnel radiation exposures. The project engineering design bases, plans, and schedule were presented to NRC staff. Prior to the commencement of the decontamination activity, the licensee agreed to perform an additional engineering analysis related to a hypothetical accidental fire in the containment building. Specifically, the licensee will determine how much containment pressure will be increased if there is a fire in a high integrity container (HIC) loaded with resin. It is considered necessary to show that this pressure difference would be within acceptable containment pressure range. The licensee stated that calculations performed for contingency planning for the decontamination project had assumed the containment would be intact throughout the decontamination process; however, this analysis of the over-pressure increase due to a fire had not been analyzed.

If you have any questions concerning this meeting, please contact Todd Jackson ((610) 337-5308) or Mark Roberts ((610) 337-5094) of my staff. A list of attendees is enclosed. The licensee distributed briefing packages for the safety classification, spent fuel pool island, and chemical decontamination discussions, which are also enclosed.

Your cooperation with us is appreciated.

Sincerely,

ORIGINAL SIGNED BY:
Craig Z. Gordon
Ronald R. Bellamy, Ph.D., Chief
Decommissioning & Laboratory Branch
Division of Nuclear Materials Safety

Enclosures:
1. Attendance List
2. Briefing Package - Safety Classification
3. Briefing Package - Spent Fuel Pool Island
4. Briefing Package - Chemical Decontamination

cc w/enclosure 1:
M. Meisner, Vice President, Nuclear Safety & Regulatory Affairs
R. Fraser, Director - Engineering
J. M. Block, Attorney at Law
P. L. Anderson, Project Manager (Yankee Atomic Electric Company)
L. Diehl, Manager of Public and Governmental Affairs
T. Dignan, Attorney (Ropes and Gray)
G. Zinke, Manager, Regulatory Affairs
W. Odell, Director, Operations
M. Ferri, Director, Decommissioning
M. Lynch, Esquire, MYAPC
P. Dostie, State Nuclear Safety Inspector
P. Brann, Assistant Attorney General
U. Vanags, State Nuclear Safety Advisor
C. Brinkman, Combustion Engineering, Inc.
W. D. Meinert, Nuclear Engineer
First Selectmen of Wiscasset
Maine State Planning Officer - Nuclear Safety Advisor
State of Maine, SLO Designee
State Planning Officer - Executive Department
F. Lavallee, State of Maine
Friends of the Coast

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ENCLOSURE 1

MEETING ATTENDANCE - MEETING NO. 98-02
JANUARY 28, 1998

Name	Title/Company
Randolph Ragland	Radiation Specialist, NRC Region I
Todd Jackson	Health Physicist, NRC Region I
Richard Rasmussen	Senior Resident Inspector, NRC Region I
Bill Raymond	Senior Resident Inspector, NRC Region I
A. Randolph Blough	Director, Division of Nuclear Materials Safety (DNMS), NRC Region I
Ronald Bellamy	Chief, Decommissioning and Laboratory Branch, NRC Region I
Mark Roberts	Senior Health Physicist, NRC Region I
Glenn Meyer	Chief, Components Engineering, Division of Reactor Safety (DRS), NRC Region I
George Pangburn	Deputy Director, DNMS, NRC Region I
Michael T. Masnik	Acting Section Chief, Decommissioning Section, Nuclear Regulatory Research (NRR), NRC
Ronald A. Burrows	Project Manager, Decommissioning, NRR
Paul J. Plante	Project Manager, Decontamination, Maine Yankee
Mike Meisner	Vice President - Nuclear Safety and Regulatory Affairs, Maine Yankee
Mark Ferri	Director of Decommissioning, Maine Yankee
Bill Hennes	Lead SFPI Engineer, Maine Yankee/Duke Engineering &Services Company (DE&S)
John Haselane	Engineering Director, Connecticut Yankee
Gerry van Noordennen	Licensing Manager, Connecticut Yankee
Patrick Dostie	State Nuclear Safety Insp., State of Maine
Jim Joosten	Partner Connect USA
Maureen Brown	Maine Yankee Public Affairs Director
Robert P. Jordan	Maine Yankee Manager, Eng. Anal.
Fred Lavallee	Maine DEP
Joan M. Jones	Maine DEP
Ray Ng	Bechtel Power Corp, 5325 Spectrum Drive, Federick, MD 21703
Bob Fraser	Engineering Director, Maine Yankee - Energy

ENCLOSURE 2

Maine Yankee

NRC Region I Briefing

January 28, 1998

Agenda

Safety classification of components	Mike Meisner
Spent fuel island design	Bob Fraser
Reactor coolant system decontamination	Paul Plante

Decommissioning

Safety Classification

January 28, 1998
NRC Region I

Safety Classification Problem/Issue

At a certain period following shutdown, strict application of regulatory requirements leads to the conclusion that no structures, systems or components (SSCs) in a decommissioning plant are safety-related (or important to safety), and therefore need not be controlled under the quality assurance program.

Safety Classification Overview

- Draft AEC Design Criterion 1: SSCs essential to prevent or mitigate accidents which could affect public safety shall be erected to quality standards that reflect their important to safety function

- 10CFR50 Appendix B: Applies to all activities affecting the safety related functions of SSCs

- 10CFR100 Appendix A: Safety-related/Important to safety definition

  - Reactor coolant pressure boundary integrity

  - Capability to shut down the reactor

  - Prevent or mitigate radiological consequences comparable to Part 100 limits

Safety Classification Decommissioning

- Underlying requirements (e.g., Appendix B, Part 100) do not change for a decommissioning plant (nor are new safety classification requirements added)

- Application/relevance of those requirements changes significantly due to:

  - Fundamental operating license changes

  - Reduced potential consequences associated with spent fuel due to reduced decay heat load

Fundamental License Changes

- Following certification of permanent cessation of operation and removal of fuel, "the 10CFR part 50 license no longer authorizes operation of the reactor or emplacement or retention of fuel into the reactor vessel" (10CFR50.82(a)(2))

- Certifications constitute both operating license and license basis changes outside of the normal 50.59/50.90 processes

- Effects on classification of SSCs devoted solely to power operation:

  - Power operation activities prohibited

  - SSCs can no longer have a safety function

  - SSCs do not meet safety-related criteria

Accident Consequence Changes

- Power operation license basis accidents (e.g., LOCA, MSLB, etc.) are no longer applicable

- Remaining accident analyses (e.g., fuel or low level waste handling events, etc.) can either:

  - Be retained as conservatively bounding events

  - Or, reanalyzed to take into account changed plant conditions (e.g., decay heat)

- New decommissioning activity events must also be addressed

SFP Events - Heatup/Boiling Analyses

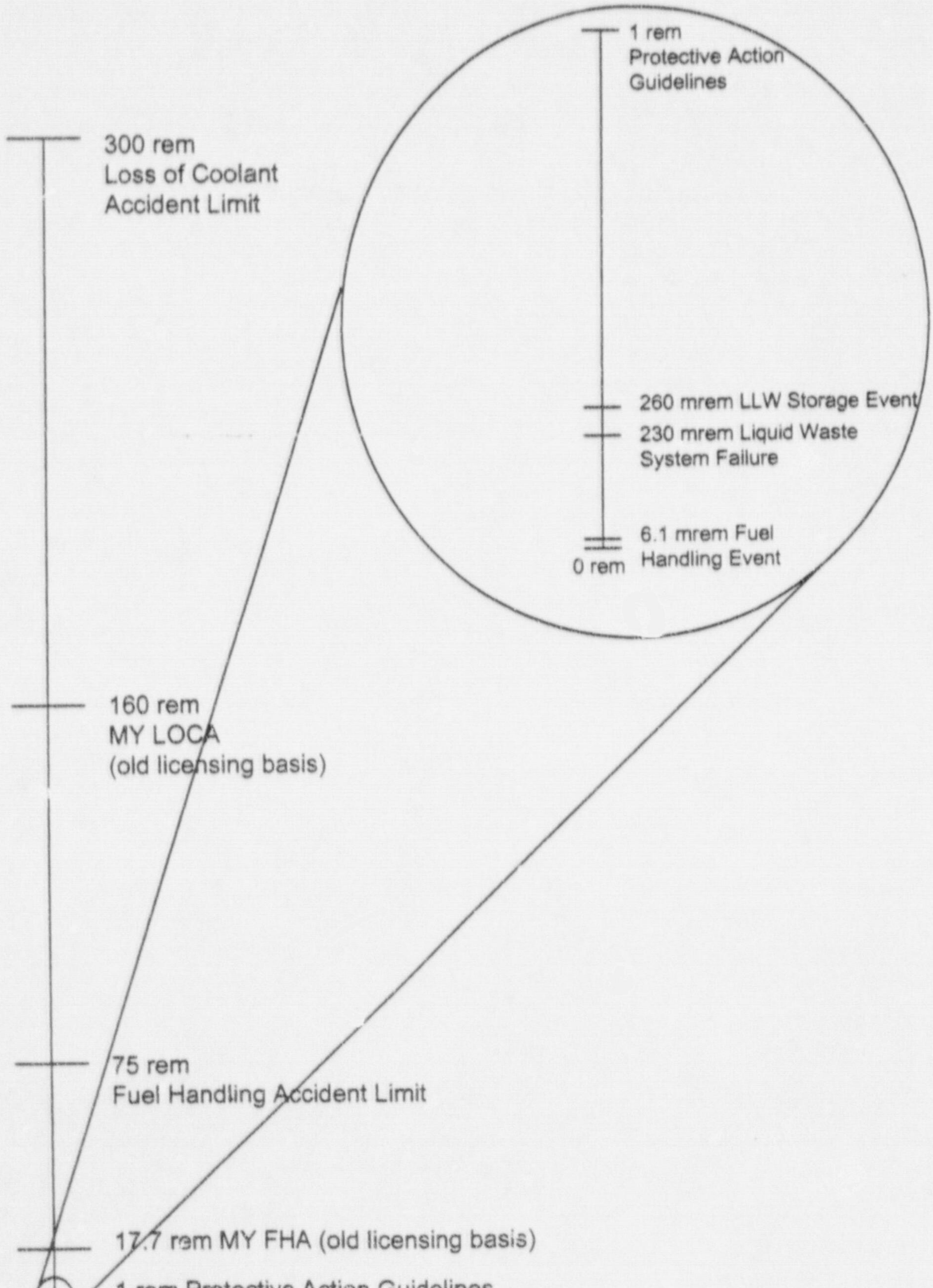
- Applicable to SFP loss of cooling and loss of inventory events
- Decay heat levels based on BTP 9.2, analyzed for 10/30/97
- License basis results:
  - Time to boil > 60 hours
  - Boil rate = 12.5 gpm (> 15 hours per foot of boiloff)
  - No offsite consequences - operator action credited
- Heatup test confirmation
  - Late October, 1997
  - Actual decay heat loads 40% less than BTP 9.2 assumptions

Events Outside of the SFP

- Liquid waste system failue
  - Primary drain tank (PDT) rupture
  - PDT at 80% capacity with undecayed RCS coolant at tech spec limit (full power FSAR analysis)
  - Calculated doses - 230 mrem (whole body)
- Low level waste storage event
  - Spill of full resin high integrity container in the LLW facility
  - 20,000 curies assumed (LLW building limit is 7,000 curies)
  - Calculated doses - 260 mrem (lung), 0.017 mrem (whole body)

Other Decommissioning Events

- Decommissioning activities
  - Removal of major components: reactor vessel, steam generators, pressurizer, etc.; RCS loop decontamination activities, etc.
- Evaluation
  - Performed in accordance with component specific activities
- Acceptance criterion
  - Included in DSAR and administratively controlled to less than limiting events (e.g., liquid waste system failure off-site doses)


1 rem
Protective Action
Guidelines
300 rem
Loss of Coolant
Accident Limit
260 mrem LLW Storage Event
230 mrem Liquid Waste
System Failure
6.1 mrem Fuel
Handling Event
0 rem
160 rem
MY LOCA
(old licensing basis)
75 rem
Fuel Handling Accident Limit
17.7 rem MY FHA (old licensing basis)
1 rem Protective Action Guidelines

Safety Classification Decomm. Events

- Limiting events
  - 230 mrem WB (PDT rupture)
  - 260 mrem lung (LLW container spill)
- More than 3 orders of magnitude below Part 100 limits (i.e., not comparable to Part 100)
- No limiting events meet safety-related definition

SSC Classification

For SSCs devoted solely to power operation:

- Implement a license basis change to establish criteria for identifying and declassifying SSCs devoted solely to power operation

- Declassify such SSCs based upon those criteria and eliminate from the license basis

SSC Classification

For SSCs associated with fuel or radiation protection functions:

- Establish the decommissioning license basis
    - Decommissioning accident analyses (confirms that consequences are not comparable to Part 100 limits)
    - Create a new classification - Important to the Defueled Condition (ITDC) - ITDC components receive augmented quality assurance
- Reclassify SSCs
    - Components previously "safety-related" become ITDC
    - Components previously non-nuclear safety remain NNS

ITDC Criteria

- Is the SSC associated with storage, control or maintenance of nuclear fuel in a safe condition; or handling of radioactive waste? This includes direct as well as indirect effects.[1]

- Is the SSC associated with radiological safety?

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

- Does the SSC satisfy a requirement based in regulations? This includes any SSC which is independently required by Technical Specification.

---

[1] Includes SSCs: 1) required to safely store and handle radioactive waste and spent fuel, 2) required to protect workers and the public from the consequences of the remaining (or new) DBAs, 3) required to safely store new fuel until it is shipped offsite, and 4) which monitor or control radiological effluent release paths.

Maine Yankee Spent Fuel Pool Island (SFPI)

Overview

- Island concept
- SFPI Project Goals
- Current SFP Cooling
- Spent Fuel Pool Island Description
- Existing SFP Monitoring
- SFPI Monitoring
- SFPI Engineered Quality Attributes
- Project Resources
- SFPI Supplemental Projects
- SFPI Project Summary

"Island" Concept

- Protect equipment that is essential to spent fuel cooling from the decommissioning activities
- Isolate non-essential equipment from spent fuel cooling systems
- Increase worker safety by removing energized systems from the decommissioning areas

SFPI Project Goals

- Establish a stand-alone, protected SFP cooling system with stand-alone support systems
- Sever plant ties to the river as the ultimate heat sink for the spent fuel pool

SFPI Project Goals (cont.)

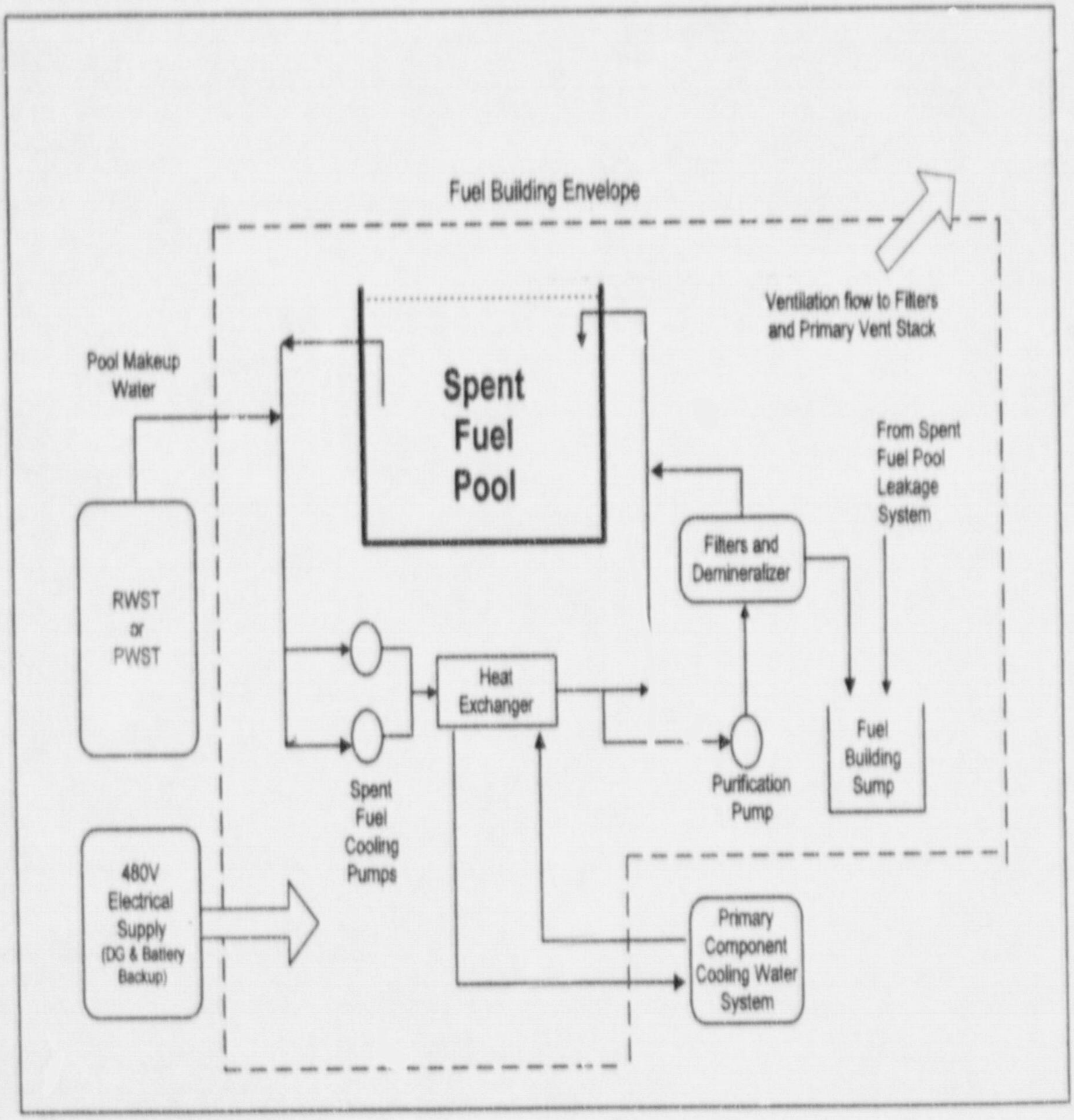
- Sever plant ties to the existing off-site power sources

- Implement the project early in the decommissioning process to enhance nuclear and worker safety

Current SFP Cooling

- Service water
- Primary component cooling
- Spent fuel cooling system
- SFP makeup and chemistry control
- Offsite power and diesel generators
- Primary auxiliary building ventilation
- Control room and security stations

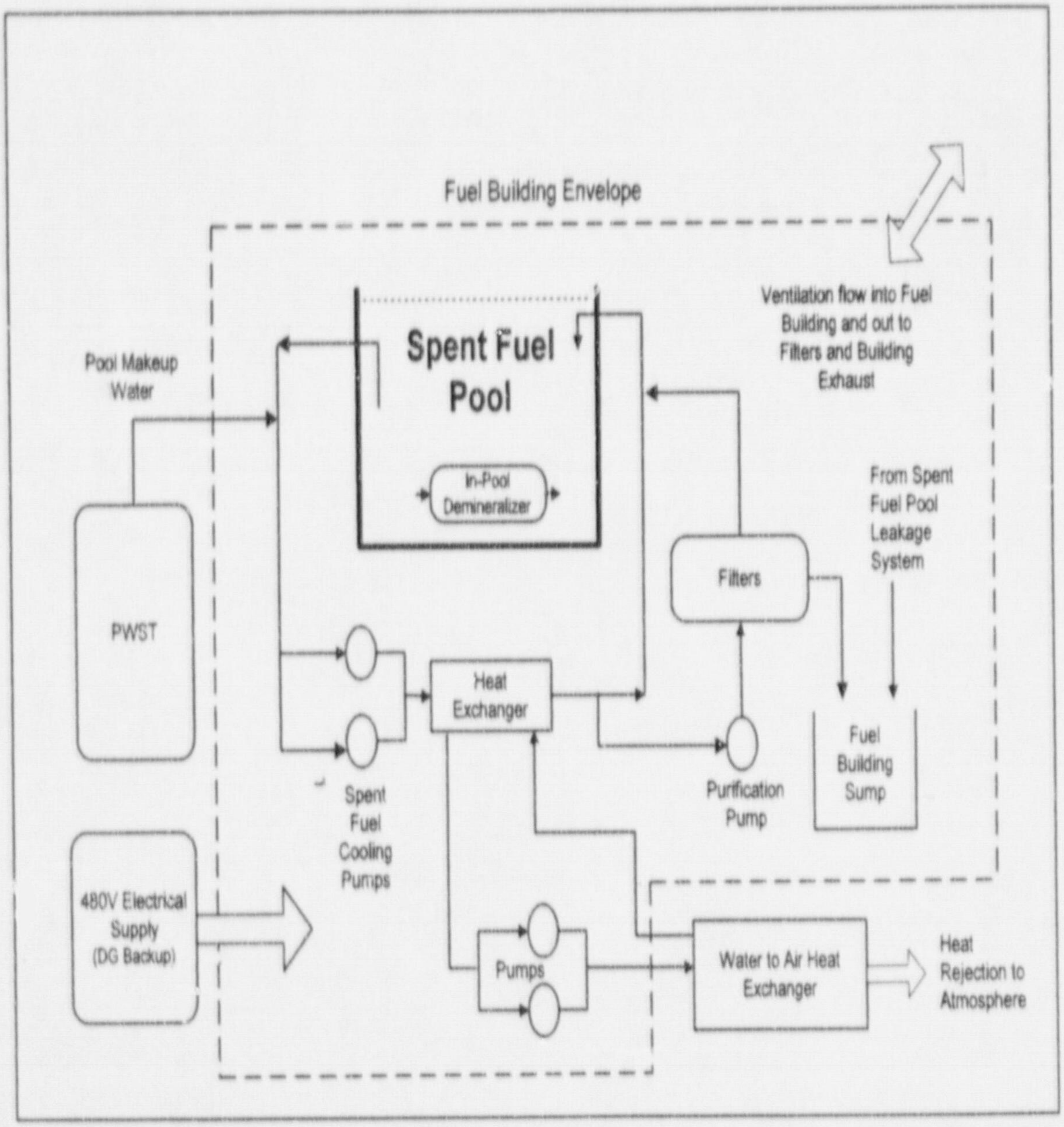
Simplified Fuel Storage Schematic



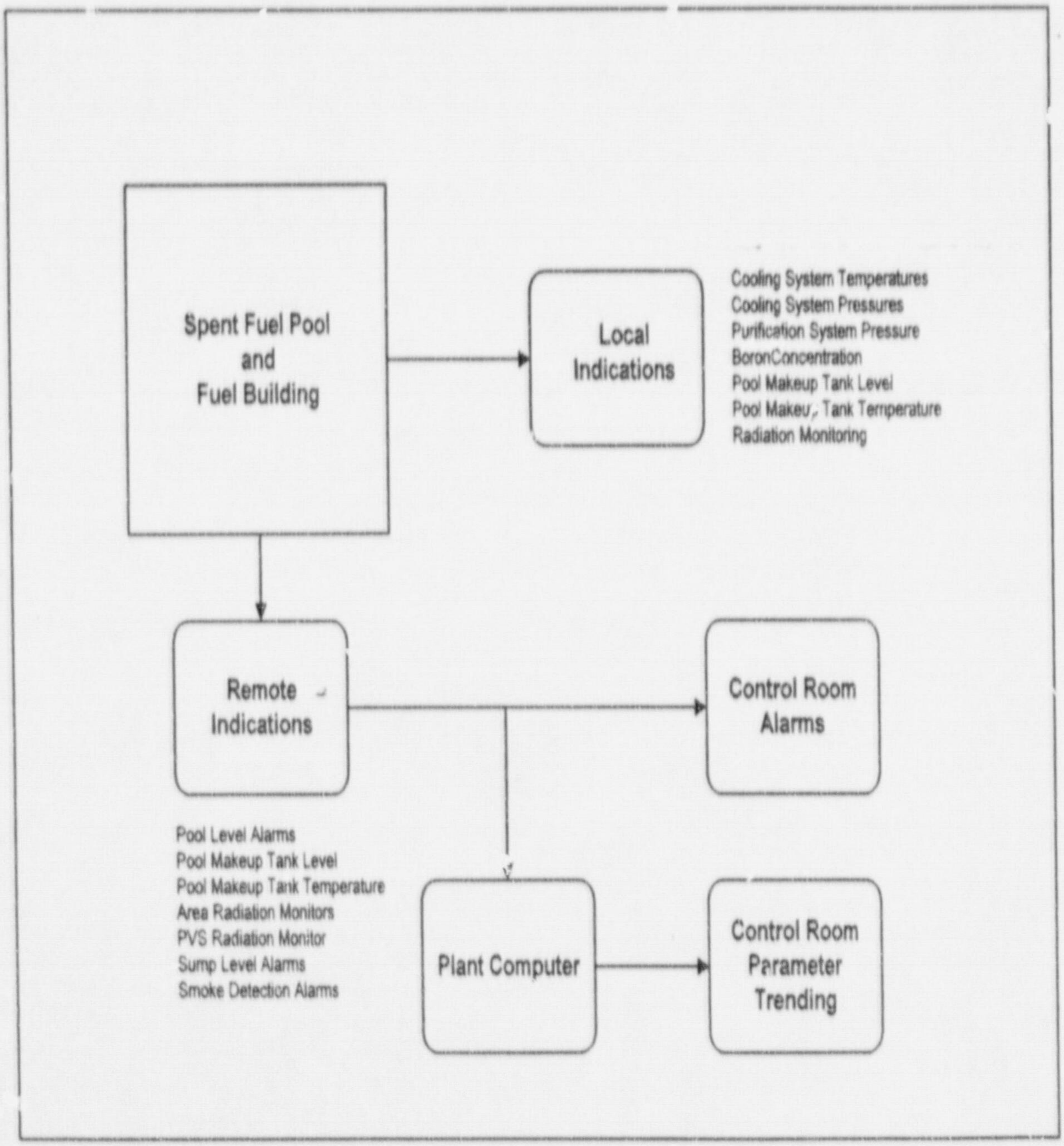
Spent Fuel Pool Island Description

- Forced air cooling for the spent fuel
- SFP purification and chemistry control
- SFP transfer tube isolation
- Dedicated power supply with backup diesel generator
- Replacement control room and security control point
- Self-contained ventilation with appropriate radiation monitors and sampling
- Security boundary, down-sized to SFPI

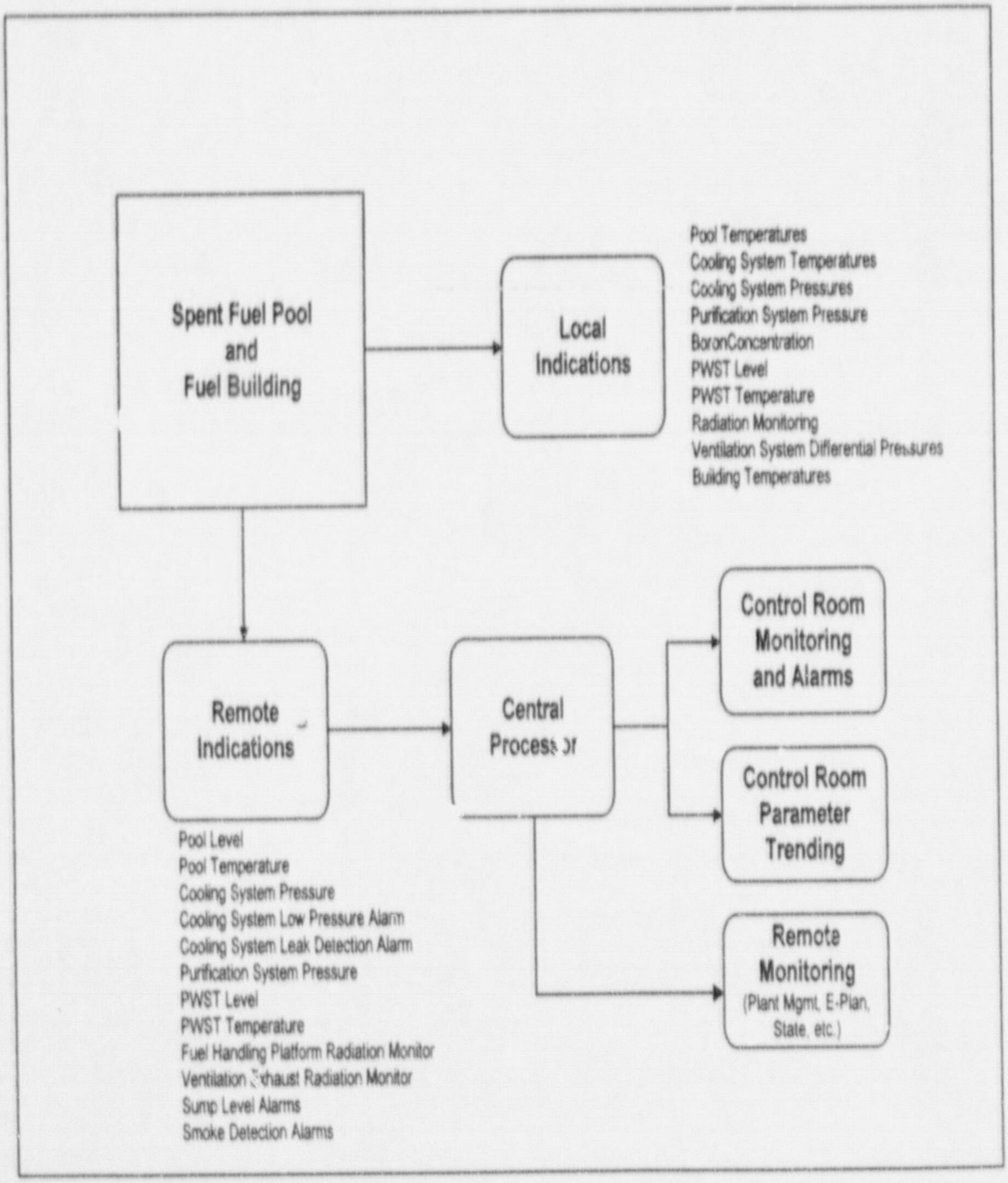
Simplified SFPI Schematic



Existing SFP Monitoring



SFPI Monitoring



SFPI Engineered Quality Attributes

- **System Design Margin**
  - Installed backup active components
  - Seismic design
  - Enhanced monitoring
  - SFP heat-up test

- **Project Controls**
  - Vendor source visits
  - Design change process
  - Installation procedures
  - Engineering oversight
  - Configuration verification

SFPI Engineered Quality Attributes

- In-process testing
  - Weld inspection
  - Hydrostatic testing
  - Concrete strength
  - QC inspections
  - Engineering inspection
  - Dimensional verification
- Pre-operational Testing
  - Thermal/hydraulic performance verification
  - Mechanical (vibration, etc.)
  - Electrical (Test tripping, DG testing, etc.)
  - System logic verification
  - Integrated system testing

SFPI Engineered Quality Attributes

- **Post Installation Surveillance**
  - Surveillance testing
  - Preventive maintenance
  - Maintenance Rule scope

- **On-going Quality Attributes**
  - Maintenance procedural controls
  - Procurement controls
  - Configuration control
  - Design control
  - Corrective action control

Project Resources

- In house engineering utilized for design

- Sandia Labs - security modifications

- Installation oversight by design engineers

SFPI Schedule

- Mechanical design change package complete 12/97
- Electrical design complete 2/98
- Installation commenced 12/97
- Materials delivery complete 3/98
- Installation complete 4/98
- Turnover to Operations 5/98
- Project close-out 5/98

SFPI Supplemental Projects

- **Control room transition project**
- **Radiation monitoring transition project**
- **Mechanical site utilities**
- **Electrical site utilities**

SFPI Project Summary

- Stand-alone, protected SFP cooling system being established
- SFP cooling will not rely on the river as the ultimate heat sink
- Dedicated power source and diesel generator for SFP cooling
- Early project implementation maximizes nuclear and worker safety

ENCLOSURE 4

Decontamination for Decommissioning

Presentation to NRC Region 1

1/28/98

Paul Plante
Project Manager
(207) 882-5806

Decontamination for Decommissioning (DfD)

A recently developed decontamination process that combines the operational advantages of dilute, traditional decontamination processes used in operating plants with the high decontamination factors achievable with concentrated aggressive processes.

or

DfD consists of circulating chemicals diluted by demineralized water through MY's piping systems to dissolve radioactivity off these pipe surfaces. The solution is then circulated to a series of filters and resin columns that will remove the radioactivity, leaving only demineralized water.

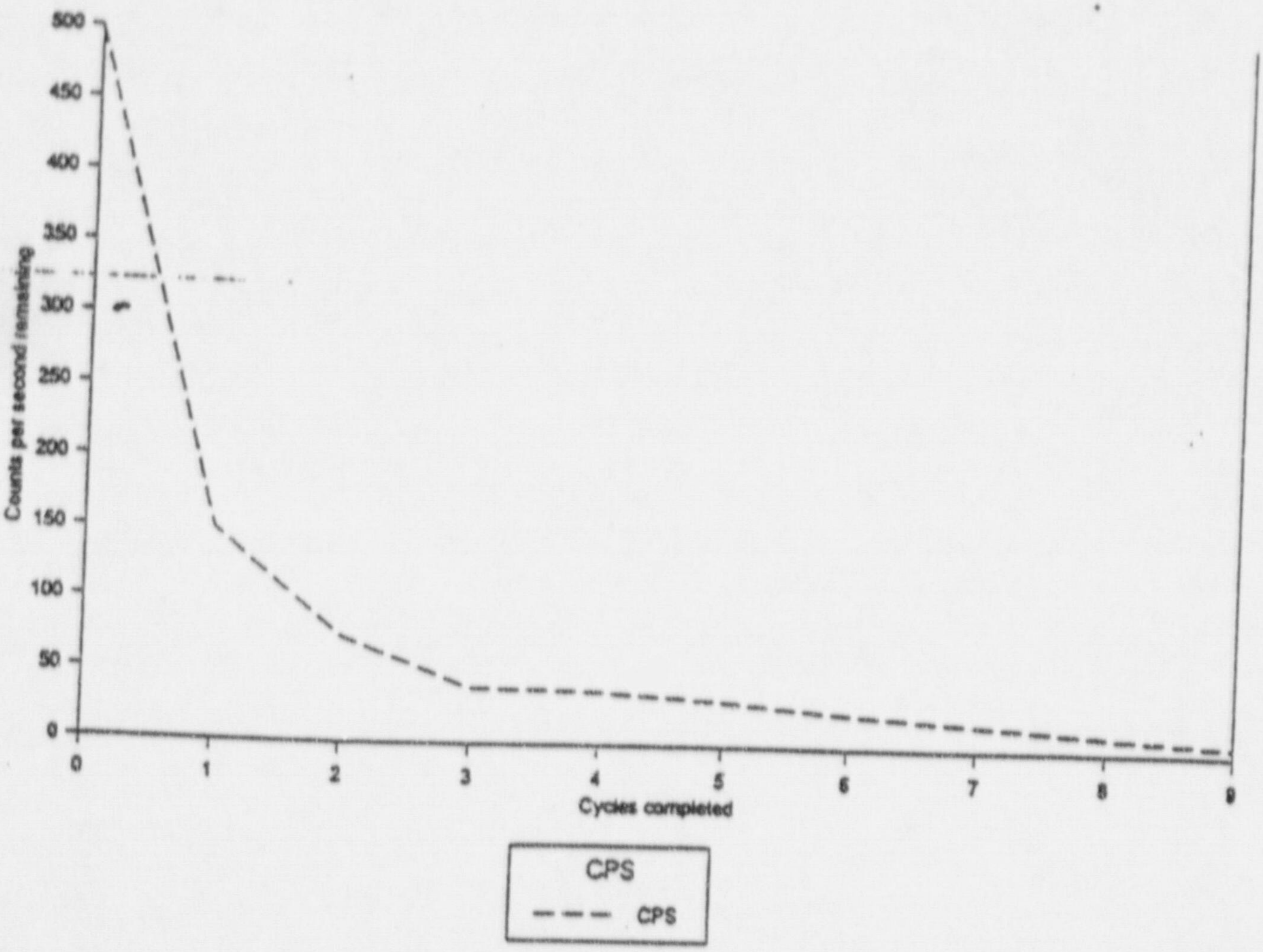
The Reasons for Chemically Decontaminating Primary Systems Now

- Improve worker safety by lowering doses & total activity levels.
- Eliminate many locked high radiation areas.
- DfD now means dose savings to everyone who works around these system in the future.
- Ensures decommissioning will not exceed the GEIS's 1115 rem limit.
- Allows DOC to proceed unimpeded with dismantling.

Decommissioning
Operations
Contractor

Schematic of the EPRI DFD Process

Step	Process	Description
1	Addition of fluoroboric acid (HBF_4) Circulation of the dilute HBF_4 acid	Concentrated HBF4 (48% to/v) mixed with system demineralized water Dissolution of iron oxides and nickel/iron oxides
2	Circulation of dilute HBF_4 with on-line ion exchange	Dissolution of iron oxides and nickel/iron oxides Removal of active and non-active cations Fe, Ni, Co and regeneration of HBF
3	Circulation of HBF_4 with addition of permanganate (100 ppm) No ion exchange	Dissolution of chrome oxides, precipitation of MnO2.
4	Circulation of dilute HBF_4 with additon of oxalic acid	Dissolution of MnO2 accompanied by the simultaneous destruction of oxalic acid. Waste products are Co2 and H 2O. CO_2
5	Destruction of any excess oxalate with potassium permanganate/cation exchange	Fine adjustment of system to achieve chemical balance Removal of active and non-active cations Fe, Ni, Co and regeneration HBF4. Removal of Mn and K.
	Repeat cycle 8 - 10 cycles	
6	On-line ion exchange cation and anion treatment	Removes all cations and anions to return the circuit to demineralized water. HBF4 anion exchg



Stainless Steel SG Tube Decontamination using EPRI DFD Solvent

Decontamination Factor (DF) / Dose Reduction Calculation

DF	% Dose Reduction (1-(1/DF))
15	93.3%
100	99.0%

- Target overall DF = 100.
- Some areas will be higher, some lower.
- Official DF = Avg. of 50 points in DfD boundary.
- Based on contact readings.

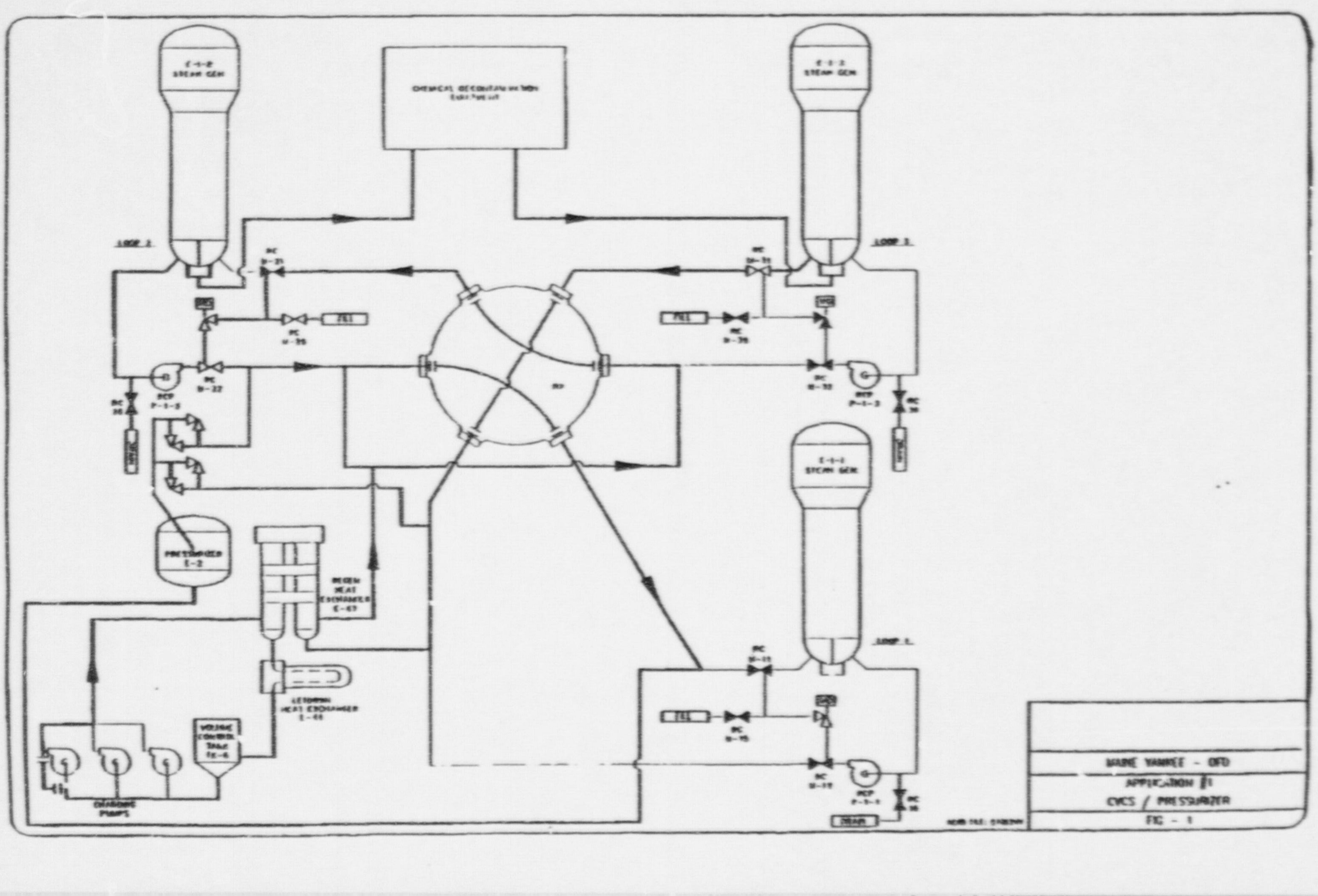
Maine Yankee's DfD Program

- First Application includes portions of the following systems:

  Charging
  Pressurizer Spray
  Letdown
  High Pressure Safety Injection
  Low Pressure Safety Injection
  Seal Water Return
  Reactor Coolant System – some parts
  Loop Drain and Fill – some parts

- Second Application includes portions of the following systems:

  Reactor Coolant System
  Loop Drain and Fill
  Residual Heat Removal

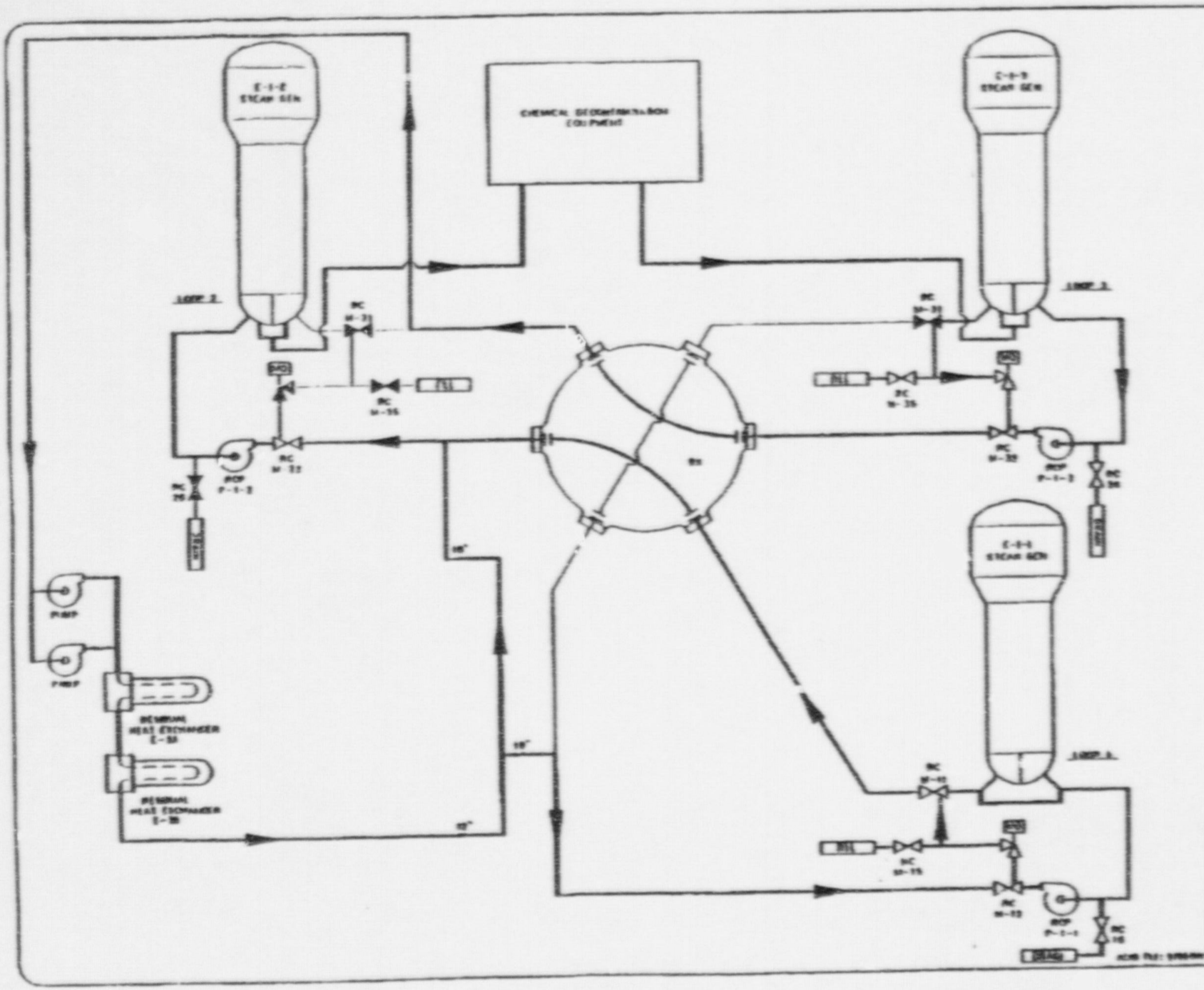


MAINE YANKEE - OFD

APPLICATION #1

CVCS / PRESSURIZER

FIG - 1

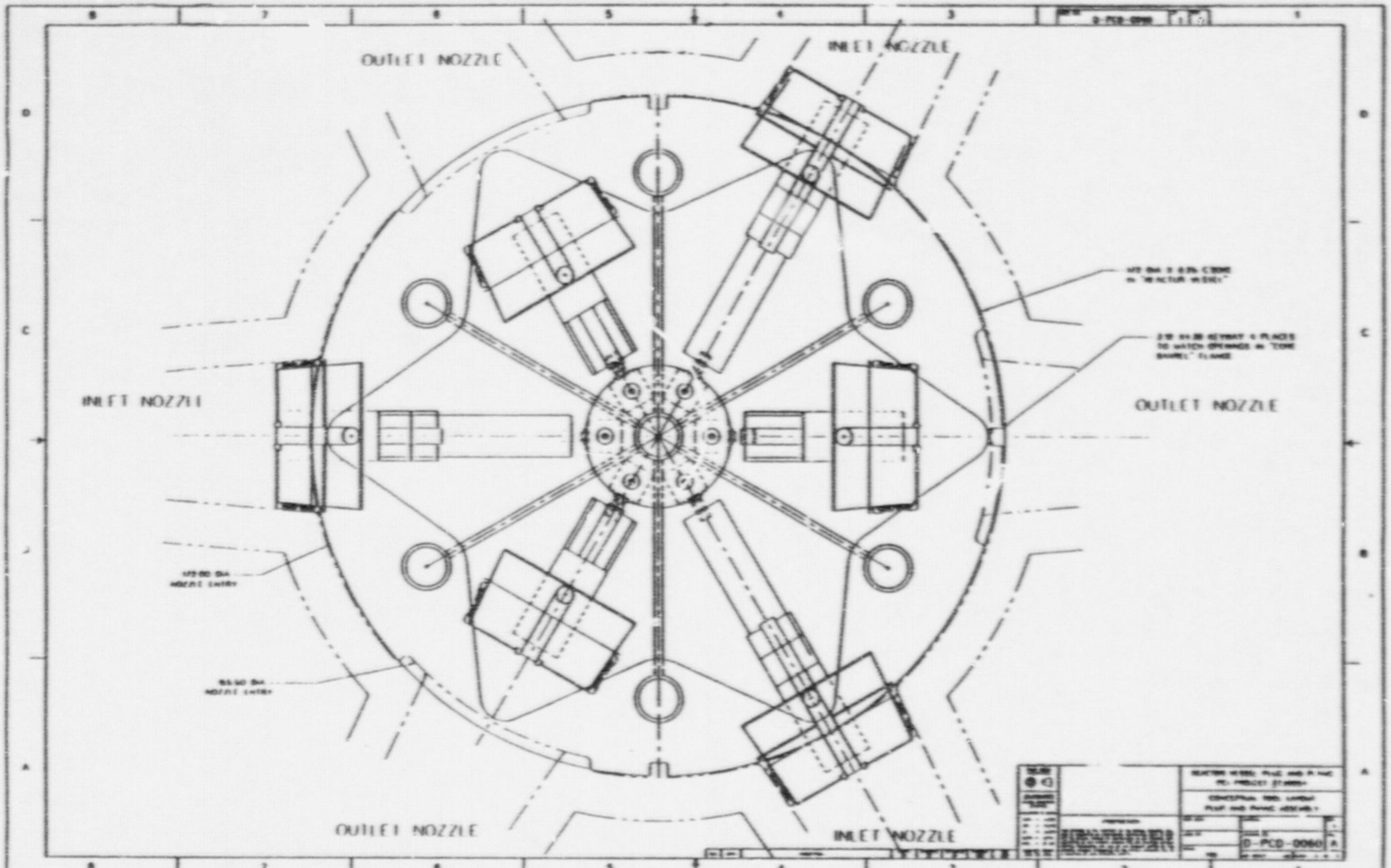


MAINE YANKEE - DFD

APPLICATION #2

RHR / LOOPS

FIG - 2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Waste Processing and Disposal

- Resin columns sluiced one at a time to a HIC inside containment.

- Each HIC dewatered inside containment by drying process addressed by Topical Report TP-02-P.

  Drying air circulated to HIC.
  Moisture removed by a chiller/water separator.
  High air temp interlocks & auto shutdown.

- Less than 600 ft^3 of resin expected from the DfD program.

  DfD Resins meet Class B disposal criteria
  Resins not mixed waste

DfD DF Survey Locations: Loop 1

RCS

Point	Location	Pre-Decon	Post-Decon	DF
1	RCP mezzanine level on RCP volute			
2	S/G platform on cold leg 33 " S/G nozzle			
3	-2' elev on 33" line between S/G and RCP			
4	19' mez. platform on hot leg 33 " S/G nozzle before RC-M-11			
5	8" Loop bypass line between RC-M-13 and RC-M-11			
6	8" Loop bypass line between RC-M-13 and RC-M-12			

Fill and Drain

Point	Location	Pre-Decon	Post-Decon	DF
7	-2 elev. 2" Drain line between 33" RCS and RC-16			
8	2" drain header after RC-M-14 from 8" bypass			
9	2" fill header after RC-M-15 from 8" bypass			
10	2" drain line after DR-12 before DR-M-3			
11	2" line after CH-M-75 heading away from HP drain cooler			
12	2" drain line between DR-A-6 and DR-A-10			

Letdown

Point	Location	Pre-Decon	Post-Decon	DF
13	2.5" letdown line after LD-1 away from PZR Cubicle.			
14	2.5" letdown line after LD-74 towards PZR Cubicle.			

HSI

	Location	Pre-Decon	Post-Decon	DF
15	underside of 10" line near HSI-16			

5-7 Hrs based on [illegible] loading or a [illegible]

Maine Yankee
RELIABLE ELECTRICITY SINCE 1972

MAIN YANKE GENERAL SURV Y RECORD FORM

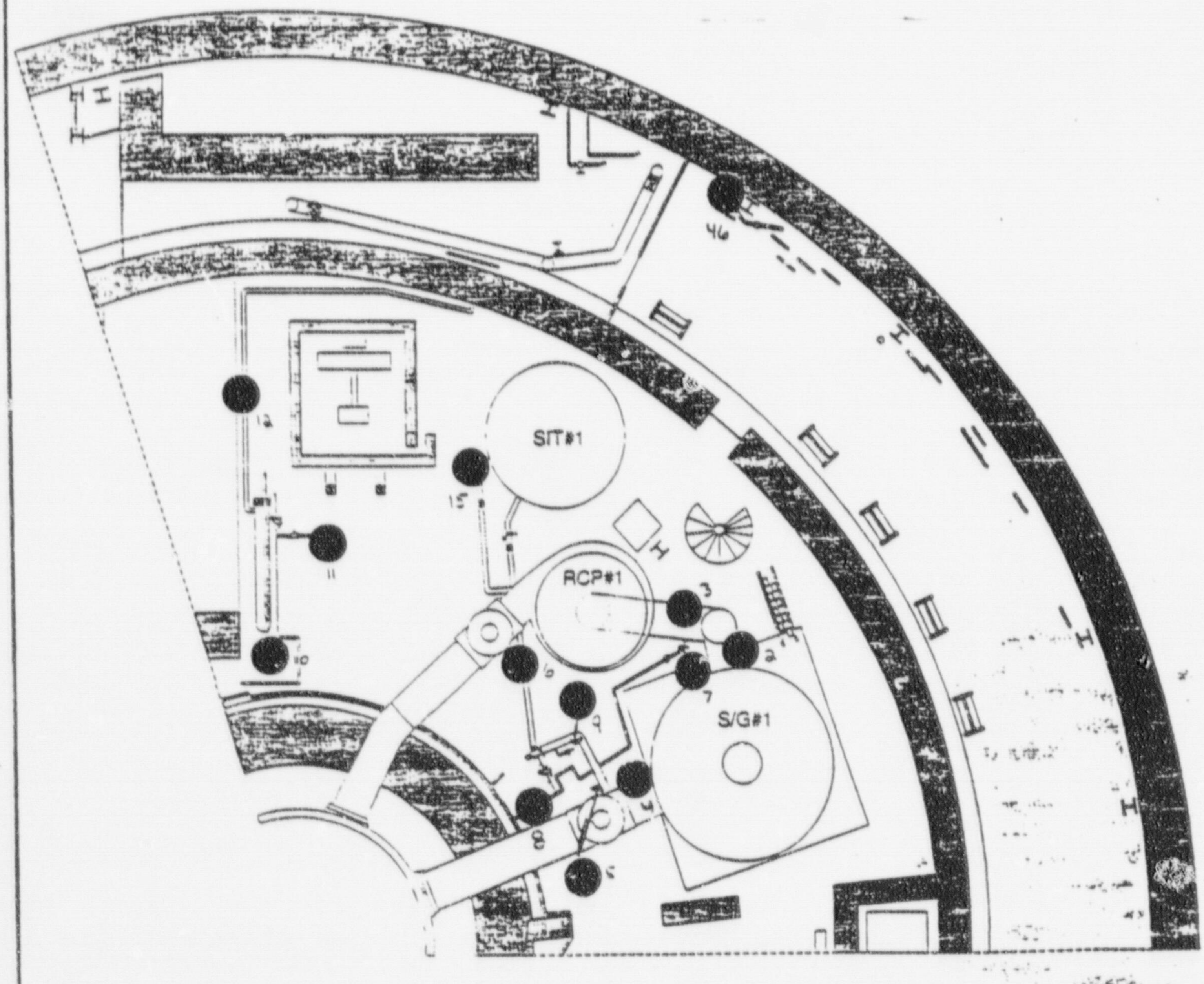
N

Map#: CB1-009	Date:	Time:	Reactor Pwr %	Tech File Number: 19.20.5.5.7	RWP's Used:	Dose Received: mR
Revision#: 00						

Surveyor Name: (Printed)	Surveyor Name: (Signature)	Location/Job Description:
		Containment Building Elevation -2'-0" - Outer Annulus - NE General Area

Required R.P. Review / Date	Required ALARA Supervisor Review / Date	REASON FOR SURVEY
		☐ ROUTINE ☐ JOB-COVERAGE ☐ SHIELDING ☐ OTHER (Specify):

INSTRUMENTS USED				CONTAMINATION RESULTS								KEY:
MODEL	SERIAL #	CAL DUE	HOA	SAMPLE #	RESULTS	SAMPLE #	RESULTS	SAMPLE #	RESULTS	SAMPLE #	RESULTS	
												• Contact exposure rates denoted by: •
												• Smear locations denoted by: ②
												• Boundaries or barriers denoted by: -x-x-
												• Dose rates denoted by:
												• Large area smears denoted by:
												• Air sample location denoted by:
												Sample Continuation Sheet Used. ☐ YES



Controls and Safety Features

Procedural:

- MY Ops procedure to lineup systems. Lineups verified.
- MY approved Vendor procedures:
  Equipment Setup, Test & Removal
  Ops procedure for each DfD program
  Resin Loading & Sluicing
  Resin Dewatering
- MY approved Vendor procedure for off-normal ops:
  Equipment leaks
  Loss of Air
  Loss of Water
  Loss of Power
  Equipment Failure
  Bulk Chemical Spills

- Maine Yankee's formal Spill Plan

Controls and Safety Features

Process:

- In process chemistry samples.
- System overpressure protection.
- Equipment pressure test.
- Predecon System Leak Test using demin water.
- Ground fault protection.
- Major runs hard piped for equipment and process reliability
- Air operated backup systems

Controls and Safety Features

Vendor Oversight & Controls:

- MY Project Manager
- MY Shift Engineers
- Dedicated Rad. Protection technicians
- Cognizant on shift Operator
- Dedicated ALARA Engineer
- Dedicated Ops Shift Supervisor
- QA Oversight

Safety Evaluation

Approach:

- ✓ **Evaluate using 10CFR50.59**

- ✓ **Decontamination Activities Assessed to FSAR Accidents:**

  1. Low Level Waste Building Contaminated Resin Spill

     Off-Site Dose: 260 mRem (Lung)

  2. Radioactive Waste System Leaks and Failures

     Off-Site Dose: 230 mRem (Whole Body)

- ✓ **Determination of 2 Hour Off-Site Boundary Dose**

Safety Evaluation

Methodology and Results:

✓ **RCS Decontamination Resin Spill**

- Secured containment
- Conservatively assume full HIC is spilled
- Conservatively assume a fire initiates airborne releases
- Airborne Releases per guidance in SAND87-2808 & NUREG/CR-0130

✓ **Preliminary Analysis Shows Off-Site Doses Well Less Than 260 mRem**

Safety Evaluation

Methodology and Results:

✓ **Contaminated Decontamination System Water Spill**

- Liquid spills contained-No outside runs of pipes. All systems in Containment, PAB, & Spray Bldg.
- Spill plan & recovery actions in place
- Decontamination water result in doses well below FSAR dose limits.

✓ **No further evaluation required. Bounded by existing analysis.**

Lessons Learned: Previous Utility DfD

✓ Loss of inventory/ spills. Spill at previous DfD due to decon fluid leaking by a boundary valve and corroding a carbon steel probe.

- Comprehensive tagouts at MY define a stainless steel, treated boundary only.
- Extensive system lineup eliminates carbon steel pathways during decon.
- At MY, systems beyond boundary valves are stainless steel.
- MY to monitor level and inventory. Aggressively investigate any unexplained inventory losses.
- If a spill occurs, MY has an off-normal ops procedure and Maine Yankee's Spill Plan to minimize and remediate.

Lessons Learned: Previous Utility DfD

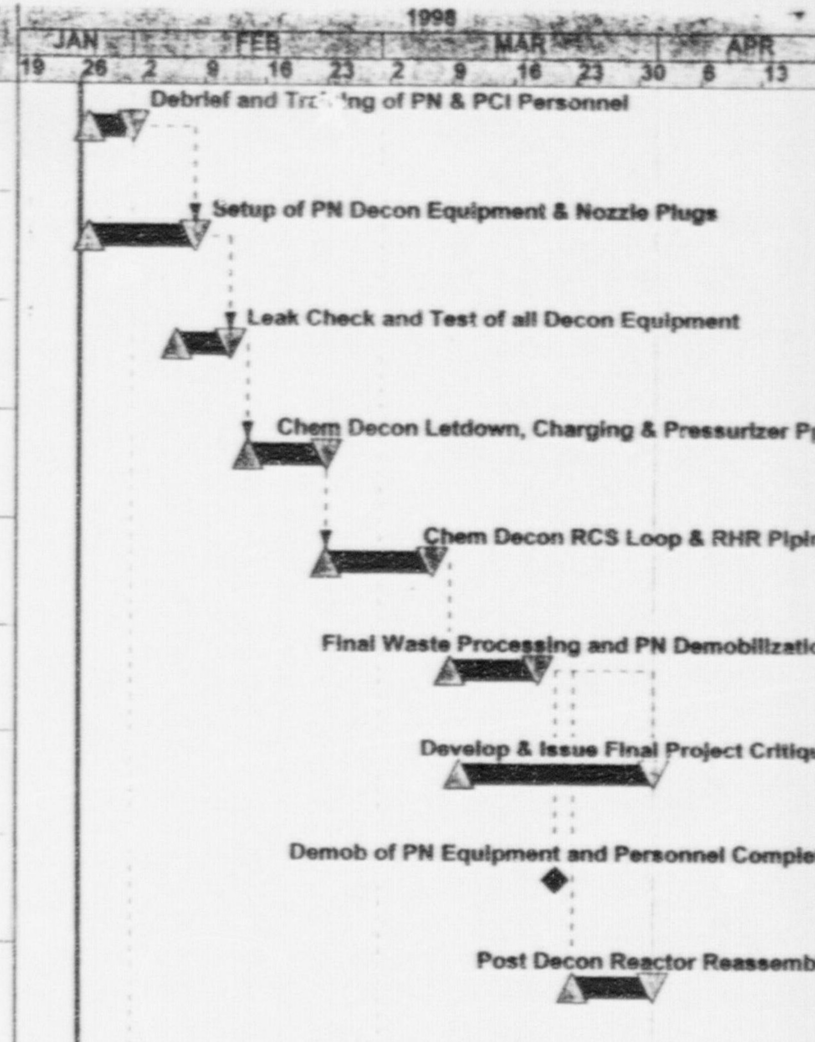
- ✓ Loss of power incident. Concern with previous DfD & loss of power was governed by carbon steel corrosion.

  - Need a loss of power response plan.
  - MY has no known carbon steel in the system to be treated.
  - MY can experience a loss of power with no adverse consequences.
  - MY has a loss of power response plan in the off-normal ops procedure.

Maine Yankee's Team Decon is Experienced, PN Services is Experienced

Project Mgr:	Managed RCP Decons in '87 & '88. Managed Loop Decon in '95.
Project Eng:	Same Engineer Loop Decon in '95.
ALARA Eng:	Same ALARA Eng Loop Decon in '95.
Shift Eng:	Same Engineer Loop Decon in '95.
Rad Con Super:	Lead Tech: Loop Decon in '95.
Rad. Waste:	Same Staff Loop Decon in '95.
Vendor:	Same Vendor from Loop Decon in '95. IP-3 full system decontamination. Big Rock DfD Decon. Chemists who developed DfD process. Same Craft Foreman as '95 Decon.

Activity ID	Activity Description	Early Start	Early Finish
PSD-001	Debrief and Training of PN & PCI Personnel	27JAN98*	31JAN98
PSD-002	Setup of PN Decon Equipment & Nozzle Plugs	27JAN98*	07FEB98
PSD-003	Leak Check and Test of all Decon Equipment	06FEB98*	11FEB98
PSD-004	Chem Decon Letdown, Charging & Pressurizer Ppg	14FEB98*	22FEB98
PSD-005	Chem Decon RCS Loop & RHR Piping	23FEB98*	06MAR98
PSD-006	Final Waste Processing and PN Demobilization	09MAR98*	18MAR98
PSD-007	Develop & Issue Final Project Critique	10MAR98*	[illegible]MAR98
PSD-008	Demob of PN Equipment and Personnel Complete		20[illegible]98*
PSD-009	Post Decon Reactor Reassembly	23MAR98[illegible]	[illegible]31MAR98



Project Start	26JAN98
Project Finish	31MAR98
Data Date	26JAN98
Run Date	26JAN98

Early Bar
Progress Bar

NRCP

Sheet 1 of 1

Maine Yankee Atomic Power Company
Primary System Decon (Level 1 Sched)
Project Manager - Paul Plante

Prepared By: J. Keto @ ext 4942

Date	Revision	Checked	Approved