STP-WI-GC

JAN 29 1988

Wisconsin Division of
Emergency Government
ATTN: R. I. Braund
Administrator
4802 Sheboygan Avenue
Madison, WI 53702

Dear Richard:

Enclosed you will find copies of three documents relating to the decommissioning of the LaCrosse Boiling Water Reactor (LACBWR). The documents are the LACBWR Decommissioning Plan, the Preliminary DECON Plan, and a Supplement to the Environmental Report for the Post-Operating License Stage - SAFSTOR. The NRC is presently reviewing these documents and will prepare a Safety Evaluation Report. I will forward a copy of this document to you when it is completed.

I am also willing to set up a meeting with State officials in Madison to discuss the decommissioning project. If there is interest in such a meeting, please contact me at 312/790-5666 to arrange a mutually agreeable time and location. I hope the information provided is useful to you. NRC welcomes any comments you may have on the project and if I can be of any assistance in this matter please contact this office.

Sincerely,

Roland Lickus, Chief
State and Government Affairs

Enclosures: As stated

cc w/enclosures:
T. Vierima, WI Rad Protection Council
L. McDonnell, WI Rad Protection Section
P. Osborne, Governor's Office
J. Hinds, MN Environmental Quality Board

cc w/o enclosures:
C. Kammerer, SLITP
P. Erickson, NRR

RIII	RIII	RIII	RIII	RIII
Lickus/kst	Jackiw	Guldemond	Paperiello	Davis
1/27/88	1/28/88	1/28/88	1/28/88	1/28/88

IE45
1/1

DAIRYLAND POWER COOPERATIVE • P O BOX 817 • 2615 EAST AVE SO • LA CROSSE, WISCONSIN 54602-0817
(608) 788-4000

JAMES W. TAYLOR
General Manager

December 21, 1987

In reply, please
refer to LAC-12460

DOCKET NO. 50-409

PRIORITY ROUTING

orig

FILE

Acknowledge receipt w/ Decommissioning Plan to DRP

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Gentlemen:

SUBJECT: Dairyland Power Cooperative
La Crosse Boiling Water Reactor (LACBWR)
Provisional Operating License DPR-45
Decommissioning Plan, Preliminary DECON Plan,
and Supplement to Environmental Report

REFERENCES: (1) DPC Letter, Taylor to Docket Control Desk, LAC-12234, dated May 22, 1987
(2) NRC Letter, Berkow to Taylor, dated August 4, 1987
(3) 10 CFR 170.12

The La Crosse Boiling Water Reactor (LACBWR) was permanently shut down on April 30, 1987. Reactor defueling was completed June 11, 1987. Reference 1 contained a request to amend the plant's license to a possession-only license. The NRC issued the possession-only license in Reference 2.

Enclosed for your review are 6 copies each of the La Crosse Boiling Water Reactor Decommissioning Plan, Preliminary DECON Plan, and Supplement to the Environmental Report for the Post-Operating License Stage - SAFSTOR. The Decommissioning Plan and Preliminary DECON Plan are bound together, the Environmental Report Supplement is separate. It is Dairyland Power Cooperative's understanding that the plant officially enters the SAFSTOR condition upon NRC approval of the Decommissioning Plan.

The Decommissioning Plan, Preliminary DECON Plan and Environmental Report Supplement have been reviewed by the Operations Review Committee and Safety Review Committee.

PC4

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Proposed Technical Specifications for the SAFSTOR period will be submitted separately.

The application fee required by Reference 3 is enclosed.

If you have any questions regarding these documents, please contact us.

Yours truly,

DAIRYLAND POWER COOPERATIVE

James W. Taylor

James W. Taylor, General Manager

JWT:LSG:dh

Enclosures

cc - Mr. A. Bert Davis
Regional Administrator
U. S. Nuclear Regulatory Commission
Region III

Mr. Peter B. Erickson, LACBWR Project Manager (9 copies)
Division of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission

Mr. Ken Ridgway, NRC Resident Inspector

Mr. L. L. Smith, Director
Electric and Water Bureau
Wisconsin Public Service Commission
Madison, WI 53707

D. Sherman (ANI Library)

PC4

SUPPLEMENT

TO THE

ENVIRONMENTAL REPORT

FOR THE

LA CROSSE BOILING WATER REACTOR
(LACBWR)

POST-OPERATING LICENSE STAGE

SAFSTOR

December 1987

~~8801150078~~ 38 pp

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1.0 INTRODUCTION

This is the Supplement to the Environmental Report for the La Crosse Boiling Water Reactor (LACBWR) covering the post-operating license period while the reactor is in the SAFSTOR condition. The Environmental Report was initially dated September 1972 and was submitted to the Atomic Energy Commission on December 8, 1972, in support of the planned application for full-term operating authorization for LACBWR. The Nuclear Regulatory Commission (NRC) issued the Final Environmental Statement related to operation of the La Crosse Boiling Water Reactor in April 1980 as NUREG-0191.

The La Crosse Boiling Water Reactor achieved initial criticality on July 11, 1967. On April 30, 1987, LACBWR was permanently shut down. Final defueling was completed June 11, 1987.

This supplement will address any significant environmental changes associated with the shutdown of the La Crosse Boiling Water Reactor and maintenance of the plant in a SAFSTOR condition.

The Decommissioning Plan for the La Crosse Boiling Water Reactor is being submitted to the NRC at the same time as this supplement. An additional Environmental Statement Supplement will be submitted with the detailed DECON Plan, prior to the end of the SAFSTOR period. This supplement, therefore, considers only the SAFSTOR decommissioning activities.

2.0 SELECTION OF SAFSTOR

The Nuclear Regulatory Commission (NRC) proposed rule on Decommissioning Criteria for Nuclear Facilities identifies 3 major classifications of decommissioning alternatives. They are DECON, SAFSTOR, and ENTOMB. The proposed rule defines the alternatives as follows:

> DECON is the alternative in which the equipment, structures, and portions of a facility and site containing radioactive contaminants are removed or decontaminated to a level that permits the property to be released for unrestricted use shortly after cessation of operations.
>
> SAFSTOR is the alternative in which the nuclear facility is placed and maintained in such condition that the nuclear facility can be safely stored and subsequently decontaminated (deferred decontamination) to levels that permit release for unrestricted use.
>
> ENTOMB is the alternative in which radioactive contaminants are encased in a structurally long-lived material, such as concrete. The entombed structure is appropriately maintained and continued surveillance is carried out until the radioactivity decays to a level permitting unrestricted release of the property. This alternative would be allowable for nuclear facilities contaminated with relatively short-lived radionuclides such that all contaminants would decay to levels permissible for unrestricted use within a period on the order of 100 years.

For a power reactor the choice is either DECON or SAFSTOR. Due to some of the long-lived isotopes in the reactor vessel and internals, ENTOMB, by itself, is not an allowable alternative.

The choice between SAFSTOR and DECON must be made based on a variety of factors including availability of fuel and waste disposal, land use, radiation exposure, waste volumes, economics, safety, and availability of experienced personnel. Each alternative has advantages and disadvantages. The best option for a specific plant has to be chosen based on an evaluation of the factors involved.

The overriding factor affecting the decommissioning decision for LACBWR is that the only feasible place to store the activated fuel is onsite. A federal repository is not expected to be available for about 20 years. With the fuel in the Fuel Element Storage Well, the only possible decommissioning option is SAFSTOR. Only limited decontamination and dismantling of unused systems can be performed during this period.

There are other reasons to choose the SAFSTOR alternative. The majority of piping contamination is Co-60 (5.27 yr half-life) and Fe-55 (2.7 yr half-life). If the plant is placed in SAFSTOR for 50 years, essentially all the Co-60 and Fe-55 will decay to stable elements. Less waste volume, with less radioactivity content, will be generated, and exposure to personnel performing the decontamination and dismantling activities will be significantly lower. Therefore, delayed dismantling supports the ALARA (As Low As Reasonably Achievable) goal. The reduction in dismantling dose exceeds the dose the monitoring crew receives during the SAFSTOR period.

The decommissioning cost estimate is discussed in Section 6.7 of the Decommissioning Plan. The majority of studies show that while the total cost of SAFSTOR with delayed DECON is greater than immediate DECON, the present value is less for the SAFSTOR with delayed DECON option.

The main disadvantage of delayed DECON is that the plant continues to occupy the land during the SAFSTOR period. The land cannot be released for other purposes. DPC also operates a 350 MWe coal-fired power plant on the site. Due to the presence of the coal-fired facility, DPC will continue to occupy and control the site, regardless of the nuclear plant's status. Therefore, the continued commitment of the land to LACBWR during the SAFSTOR period is not a significant disadvantage.

A second disadvantage of delaying the final decommissioning is that the people who operated the plant would not be available for the DECON period. When immediate DECON is selected, some of the experienced plant staff would be available for the dismantling. Their knowledge of plant characteristics and events could be extremely helpful. In the absence of these knowledgeable people, all information has to be obtained from plant records. When SAFSTOR is chosen, efforts must be made to maintain excellent records to compensate for the lack of staff continuity.

The remaining factor to be discussed is safety. As of August 1987, 43 power reactors have been shut down worldwide, 19 of which are in the United States. All 3 methods of decommissioning are being used. Experience has shown that all can be used safely.

A possible fourth decommissioning alternative exists which combines some features of SAFSTOR and DECON. The possibility exists to use the secondary side of the plant with a new fossil-fired steam supply system. While DPC is not planning on pursuing this option at this time, it should not be eliminated as an alternative.

After evaluating the factors involved in selecting a decommissioning alternative, Dairyland Power Cooperative decided to choose an approximately 30-50 year SAFSTOR period, followed by DECON. The exact duration of the SAFSTOR period will be dependent on the availability of the federal fuel repository, availability of waste disposal, economics, personnel exposure, and various institutional factors. If any major changes are made in DPC's decommissioning plans, a revision to this plan will be prepared.

REFERENCES

1) Nuclear Regulatory Commission, proposed rule on Decommissioning Criteria for Nuclear Facilities, Federal Register, Vol. 50, No. 28, February 11, 1985.

2) "Decommissioning - Demonstrating the Solution to a Problem for the Next Century," Nuclear Engineering International, Vol. 32, No. 399, October 1987, p. 48.

3) Proceedings from the 1987 International Decommissioning Symposium, Conf-871018, October 4-8, 1987.

3.0 ONSITE STORAGE OF FUEL

Eleven fuel cycles over the 20 years of operation have resulted in a total of 333 irradiated fuel assemblies being stored in the LACBWR Fuel Element Storage Well. Fifty-two new fuel assemblies remain in New Fuel Storage. It is expected that the new fuel will be sent offsite during 1988.

DPC plans to continue to store the irradiated fuel assemblies in the Fuel Element Storage Well (FESW). The FESW is located in the Containment Building. The storage well is an 11-foot square pool, approximately 42 feet deep. The fuel is stored in a two-tier configuration in high density storage racks.

The Nuclear Regulatory Commission issued its Waste Confidence Decision in the Federal Register on August 31, 1984. In it, the NRC found "reasonable assurance that, if necessary, spent fuel generated in any reactor can be stored safely and without significant environmental impacts for at least 30 years beyond the expiration of that reactor's operating license at that

reactor's spent fuel storage basin, or at either onsite or offsite independent spent fuel storage installations." Therefore, DPC's plan to maintain the activated fuel at LACBWR, until a federal repository is ready to accept the fuel, is acceptable from the safety standpoint, as well as necessary from the practical standpoint.

4.0 PLANT STATUS AND ACTIVITIES

The LACBWR Decommissioning Plan provides a description of the plant and discusses the status of plant systems. It also contains an estimation of the plant's radionuclide inventory.

The Decommissioning Plan describes the preparations for SAFSTOR and activities planned for the SAFSTOR period. The Decommissioning Program and its administrative controls are discussed. The Decommissioning Plan also describes the Radiation Protection Program, radioactive waste handling, and various monitoring programs. These programs are quite similar to those used during plant operation.

5.0 CHANGE IN LAND USE

During the SAFSTOR period, LACBWR will continue to occupy its present site. The land committed to the plant will remain the same; only the function of the facility will change. The site will be used for safe storage of the decommissioned nuclear plant, instead of for an electric power generating facility.

Adjacent to LACBWR is a coal-fired power plant, Genoa Unit No. 3. Due to the presence of this facility, DPC will continue to use the site for industrial purposes, regardless of the status of the nuclear plant.

5.1 Effect on Historical Sites

DPC contacted the State Historical Society of Wisconsin to inquire whether the proposed decommissioning would have any effect on any property listed in the National Register of Historic Places or eligible to be listed.

The reply stated that the first stage of decommissioning (SAFSTOR) would not have any potential to affect such historic places. The final decommissioning (DECON), however, could have effects on historical and archeological sites. Therefore, the impact on historical sites will have to be evaluated during the planning for the DECON effort.

5.2 Manpower

The number of DPC employees at the nuclear plant during the SAFSTOR period is expected to be approximately 30% of the operating staff. Traffic associated with LACBWR will be less than during plant operation.

6.0 THERMAL EFFLUENTS

The thermal release from the plant during SAFSTOR will be significantly less than that experienced during plant operation. No longer will the waste heat from the steam cycle be released to the Mississippi River. The main sources of heat remaining are the decay heat from the irradiated fuel, which will continue to decrease with time, and the heat generated by operating equipment, e.g. the air compressors.

While all open water systems listed in the Environmental Report will continue to be operated, the water use will be less. The Circulating Water System will be operated only intermittently, rather than continuously. Less water will also be needed to compensate for system leakage. Thermal discharges will be accomplished primarily by the Low Pressure Service Water System through the same thermal effluent pathway to the Mississippi River as during plant operation.

The shutdown of the 50 MWe LACBWR will not affect the thermal discharges from Genoa Unit No. 3, the 350 MWe coal-fired generating unit onsite. A common discharge into the river is used by both plants. The original Environmental Report discussed that it was unlikely that both plants would be shut down at the same time, and so the probability of subjecting the fish to cold-shock if LACBWR was shut down was small. With LACBWR permanently shut down, its thermal discharges cannot reduce the effect of a shutdown of Genoa Unit No. 3. Due to the immediate mixing at the discharge point, however, thermal shock has not been seen. Therefore, the decommissioning of LACBWR will not cause a significant greater environmental effect when Genoa Unit No. 3 shuts down.

7.0 ENVIRONMENTAL EFFECTS DURING SAFSTOR

This section describes the environmental impacts during the LACBWR SAFSTOR period. The principal potential sources of impact are gaseous and liquid releases of radioactive material, solid radioactive waste processing and shipments, chemical wastes, and noncombustible solid wastes. This section describes limits and methodology to calculate offsite doses.

Estimated doses to Members of the Public from releases of radioactive material during the SAFSTOR period are included in this section. A discussion of the comparative releases during SAFSTOR to releases during normal plant operation are also presented. The Radiological Environmental Monitoring (REM) Program description along with current and projected SAFSTOR results of the REM program are included in this section.

Effluent discharges from LACBWR will be significantly below applicable state and federal regulations during the SAFSTOR period. Estimated doses to Members of the Public from liquid and gaseous releases of radioactive materials will be significantly less during SAFSTOR than during normal plant operations, which were below the applicable 10 CFR 50 Appendix I dose limits.

The release of chemicals and hazardous materials from LACBWR during the SAFSTOR period should be insignificant. As discussed in Section 6.0, heat released by the plant's circulating water system during the SAFSTOR period will not be of any significance.

7.1 Effluent Release (10 CFR 50 Appendix I) Limits:

The Effluent Release (10 CFR 50 Appendix I) Limits which are enumerated by LACBWR Technical Specifications will still apply during the SAFSTOR period. Specifically, these required limits for effluent releases during the SAFSTOR period are:

7.1.1 Liquid Releases

Requirement #1:

"The concentration of radioactive material released in liquid effluents at any time to areas beyond the EFFLUENT RELEASE BOUNDARY shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2,...for radionuclides other than dissolved or entrained noble gases (Kr-85). For dissolved or entrained noble gases (Kr-85), the concentration shall be limited to 2×10^{-4} µCi/ml total activity."

Requirement #2:

"The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released to areas beyond the EFFLUENT RELEASE BOUNDARY shall be limited:

a. During any calendar quarter to $\leq$ 1.5 mRems to the total body and $\leq$ 5 mRems to any organ, and

b. During any calendar year to $\leq$ 3 mRems to the total body and $\leq$ 10 mRems to any organ."

7.1.2 Gaseous Releases

Requirement #1:

"The dose rate due to radioactive materials released in gaseous effluents to areas beyond the EFFLUENT RELEASE BOUNDARY shall be limited to the following:

a. The dose rate limit for noble gases shall be $\leq$ 500 mRems/yr to the total body and $\leq$ 3000 mRems/year to the skin, and

b. The dose rate limit for H-3 and for all radioactive materials in particulate form with half lives greater than 8 days shall be $\leq$ 1500 mRems/year to any organ."[1]

Requirement #2:

"The air dose to a MEMBER OF THE PUBLIC due to noble gases released in gaseous effluents to areas beyond EFFLUENT RELEASE BOUNDARY shall be limited to the following:

a. During any calendar quarter, to $\leq$ 5 mRads for gamma radiation and $\leq$ 10 mRads for beta particle radiation, and

b. During any calendar year, to $\leq$ 10 mRads for gamma radiation and $\leq$ 20 mRads for beta particle radiation."

Requirement #3:

"The dose to a MEMBER OF THE PUBLIC from H-3 and all radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents released to areas beyond EFFLUENT RELEASE BOUNDARY shall be limited to the following:

a. During any calendar quarter to $\leq$ 7.5 mRems to any organ, and

b. During any calendar year to $\leq$ 15 mRems to any organ."[1]

7.1.3 Solid Radioactive Waste Processing and Shipments

Requirement #1:

"Solid radioactive wastes shall be handled in accordance with a PROCESS CONTROL PROGRAM in order to meet shipping and burial ground requirements."

7.1.4 Radiological Environmental Monitoring (REM) Program

Requirement #1:

"The radiological environmental monitoring program shall be conducted as specified in Table 4.4.1-1. An Interlaboratory Comparison Program for annual analyses of radioactive materials shall be conducted."[2]

---

[1] A technical specification submittal to remove I-131 and I-133 from this requirement is planned since these radionuclides have decayed to stable elements and are no longer being produced at LACBWR.

[2] A technical specification submittal to remove reference to a land use census from this requirement is planned, since activity concentrations in gaseous releases have diminished significantly and an annual vegetation sample will be taken.

Cumulative doses to Members of the Public due to liquid, gaseous releases and direct radiation from the plant will be determined as required in accordance with methodology and parameters found in the LACBWR Offsite Dose Calculation Manual (ODCM). The ODCM is described in Section 7.5. The REM Program supplements the effluent release monitoring and dose assessment program by verifying through sample collection and analysis that the measurable concentrations of radioactive materials and levels of radiation released from the plant effluents are not higher than those expected on the basis of effluent measurements.

7.2 Liquid Releases

The purpose of the liquid waste processing and treatment system is to (1) collect and temporarily store radioactive liquid waste generated in the plant, and (2) to process and discharge liquid wastes in compliance with 10 CFR 50 Appendix I limits. The total fission and activation products in liquid releases from the plant are summarized in Table 7-1 for the years 1974-1984, 1985 and 1986, the period January-March 1987, and the period July-September 1987. The plant was operational during 1985, part of 1986 and for the period January-March 1987. The reactor was shut down on April 30, 1987, and reactor defueling was completed in June 1987. The liquid releases of tritium (H-3) are summarized in Table 7-2. Major radionuclide breakdowns (isotopic analyses) for 1985 through 1987 are summarized in Table 7-3.

After liquid waste is collected in a tank, the contents are recirculated in the tank using the tank transfer/recirculating pump with the required recirculation piping line-up. After a recirculation period, a sample is taken from the tank and analyzed for radioactivity concentrations and effective MPC_w (Maximum Permissible Concentration in water ratio). A batch discharge record is documented, including necessary circulating water dilution flow rates and effective MPC_w upon entrance into the Mississippi River water. The tank's contents are discharged through one of two backwashable filters and pass full flow through a shielded hemispherical stainless steel marinelli chamber, housing a sensitive radiation detector which reads out in the control room. This radiation detection system has a conservatively set alarm set point, which if reached would activate a trip which would terminate the discharge by closing the automatic liquid waste discharge flow control valve. The total discharge volume is measured by a flow totalizer. The liquid effluent discharges into the circulating water system. The discharged waste water is further diluted by mixing with the Genoa-3 circulating water discharge and with the waters of Thief Slough. The waters of the slough mix with the mainstream of the Mississippi River and effect further dilution.

Using individual batch discharge radionuclide analyses and flow rates, offsite dose calculations to Members of the Public are periodically performed. No credit is taken for further dilution in Thief Slough or the mixing zone of the mainstream of the Mississippi River in performing these calculations.

Theoretically, Members of the Public engaged in recreational activities, such as swimming or boating, could receive direct external exposure from the beta and gamma radiation emitted from the radionuclides in the water. However, the doses associated with these recreational endeavors should not exceed approximately 0.005 mRem whole body dose in any year, assuming a very conservative exposure time of 100 hours in the vicinity of the circulating water discharge and based upon the highest concentrations of radionuclides discharged in liquid waste. The dose from this pathway is not considered in calculations, since the dose is insignificant and boating and swimming do not normally take place directly at the outfall.

Theoretically, a Member of the Public could receive an internal dose from consumption of Mississippi River water which is processed for a potable water supply. However, the drinking water pathway is not considered in the offsite dose calculations, since the nearest community which obtains its drinking water supply from the Mississippi River is located in Davenport, Iowa, 195 miles downstream. The dose from this source would be immeasurable.

The only significant dose pathway used for offsite dose calculations to Members of the Public for liquid effluents is the dose commitment due to the adult ingestion of fish from the Mississippi River. These dose calculations for the adult-fish consumption pathway, very conservatively assume that the Adult Member of the Public consumes 21 kilograms/year of the edible portions of fish, which have continuously resided at the circulating water discharge outfall. Results of dose calculations for this pathway are found in Table 7-4. These dose pathways are illustrated in Figure 7.1.

After LACBWR was permanently shut down in April 1987, marked changes in the magnitude and composition of the radioactive liquid effluents were observed. The total tritium (H-3) activity discharged in the 1st quarter of 1987 decreased from 25.2 Curies to 1.47 Curies in the 3rd quarter of 1987. The H-3 activity levels discharged during the SAFSTOR period will continue to decrease due to lack of production through fission and activation of deuterium and due to radioactive decay. Cobalt-60 and Iron-55 continue to represent about 75-80% of the discharged activities and contribute the most toward calculated whole body doses. Cesium-134 and Cesium-137 represent about 5-6% of the discharged activity. They are major contributors toward organ doses. These radionuclides will continue to be the predominant activity in the liquid releases during the SAFSTOR period. Although the total activity released per calendar quarter may initially change due to variations in tasks performed at the plant, the overall trend in release activities and offsite doses from liquid discharges will decrease significantly during the SAFSTOR period. This will be due to reduced plant radioactivity concentrations because of lack of production and because of radioactive decay. Shorter-lived radionuclides, such as I-131, noble gases, Ba(La)-140 and Tc-99m, which contribute to doses during operation, were nondetectable in liquid releases during the 3rd quarter 1987. Most of these radionuclides have completely decayed to stable elements and will no longer be produced during SAFSTOR. Other radionuclides, such as Mn-54, Cr-51, Co-58 and Ru(Rh)-106, will be nondetectable in liquid discharges within a year or two, because of decay.

The activity concentrations in batches which are currently being discharged range from about 2×10^{-5} to about 6×10^{-3} µCi/ml with an average concentration of about 8×10^{-4} µCi/ml. The average batch concentrations will continue to vary initially but will decrease in the longer term SAFSTOR period.

The total volume of liquid discharges during the period January-September 1987 was 276,842 gallons. The total volume discharged from July-September 1987 was 70,253 gallons. The estimated total volume to be discharged in 1987 is 350,000 gallons. The initial SAFSTOR volumes of liquid discharges will vary, depending upon plant tasks, but should be an average of about 150,000 gallons per year. Eventually, the volumes discharged should decrease during the long-term SAFSTOR period.

Normal and potential sources of liquid wastes processed during SAFSTOR include the following:

1) Turbine Building floor drains and sumps - laundry water, laboratory water, decontamination liquids.

2) Waste Treatment Building sump - decontamination liquids, resin dewatering liquids.

3) Containment Building sumps and Retention Tanks - FESW leakage, air conditioner condensate, decontamination liquids and system draining evolutions.

TABLE 7-1

TOTAL FISSION AND ACTIVATION PRODUCT ACTIVITY
(EXCLUDING H-3)
RELEASED IN LIQUID EFFLUENTS
(Curies)

1974-1984 (avg/yr)	1985	1986	Jan-Mar 1987	July-Sep 1987
7.6 ± 6.1	1.84	5.00	0.157	0.320

TABLE 7-2

TOTAL H-3 ACTIVITY
RELEASED IN LIQUID EFFLUENTS
(Curies)

1985	1986	Jan-Mar 1987	July-Sep 1987
128.0	57.5	25.2	1.47

TABLE 7-3

MAJOR RADIONUCLIDE BREAKDOWN
IN LIQUID RELEASES
(Ci)

Radionuclide	1985	1986	Jan-Mar 1987	July-Sep 1987
Co-60	0.908	1.892	0.067	0.216
Fe-55	0.298	1.381	0.053	N.A.
Mn-54	0.115	0.516	0.014	0.045
Cs-137	0.083	0.094	0.006	0.019
I-131	0.069	0.031	<0.001	N.D.
I-133	0.059	0.072	<0.001	N.D.
Ba(La)-140	0.056	0.019	0.002	N.D.
Tc-99m	0.044	0.020	0.001	N.D.
Xe-135	0.029	0.009	<0.001	N.D.
Np-239	0.027	0.008	<0.001	N.D.
I-135	0.021	0.007	N.D.	N.D.
Co-58	0.020	0.020	<0.001	0.002
Sr-91	0.015	0.004	<0.001	N.D.
Xe-133	0.015	0.017	<0.001	N.D.
Mo-99	0.013	0.034	N.D.	N.D.
Cr-51	0.011	0.095	<0.001	<0.001
Xe-131m	0.011	0.050	N.D.	N.D.
Sr-90	0.004	0.005	<0.001	N.A.
Ru(Rh)-105	0.004	0.044	<0.001	<0.001
Ru(Rh)-106	0.004	0.009	0.001	0.002
Cs-134	0.003	<0.001	<0.001	<0.001
Sr-89	0.002	0.005	N.D.	N.A.
Ag-110m	0.002	0.017	0.008	0.014

N.D. = Not detectable above LLD.

N.A. = Not available at the time of the report.

TABLE 7-4

CALCULATED DOSES TO MEMBERS OF THE PUBLIC
FROM LIQUID EFFLUENT RELEASES
(mRem)

	1985	1986	Jan-Mar 1987	July-Sep 1987
Whole Body	0.803	2.033	0.093	0.611
Highest Organ	1.211	3.136	0.139	0.829

7.3 Gaseous Releases

The purpose of the gaseous waste holdup, filtration and treatment systems was to (1) reduce radioactivity discharged to the environment through the gaseous release systems, (2) allow for sufficient decay of shorter-lived noble gases, activation gases, and radioiodines prior to discharge to the environment, and (3) control radioactive gaseous emissions within the requirements of 10 CFR Appendix I limits.

The total fission and activation product activity released in gaseous effluents is shown in Table 7-5 for the years 1985 and 1986, the period January-March 1987, and the period July-September 1987. As previously discussed, the plant was operational during 1985, part of 1986, and for the period January-March 1987. The period July-September 1987 reflects the current conditions of LACBWR as a defuelled-nonoperational facility. The major radionuclide breakdowns (isotopic analyses) for the periods of time described are listed in Table 7-6.

During normal plant operation, the majority of the radionuclides discharged via gaseous releases to the environment were from offgas (non-condensible gases) from the condenser hotwell, which was maintained at a vacuum. The majority of these radioactive gases were noble gas radionuclides. These gases passed through offgas piping from the air ejector. To reduce their environmental impact, an augmented piping and tank system was utilized to hold up these gases to allow for sufficient radioactive decay of a number of radionuclides, which reduced the total emission rate from the plant stack. The gaseous release from the offgas system remained well within applicable federal and state limits during normal operation.

The remaining sources of radioactive gaseous releases are from various ventilation exhaust systems. These include the Containment Building, Turbine Building and the Waste Treatment Building Exhaust Ventilation Systems which, essentially, join at the stack plenum prior to discharge up the stack. One or two stack blowers, rated at about 35,000 cfm each, act as a driving force for stack releases and add dilution air to reduce the effective MPC_A within the stack prior to discharge.

The offgas system from the condenser is no longer in operation since the plant is shut down. Therefore, the majority of gaseous releases are from the various ventilation exhaust systems. The Containment Building and Waste Treatment Building Exhaust Systems are equipped with full flow HEPA filter banks. The stack currently is continuously monitored for beta particulate, radioiodine[1] and noble gas activities, with at least one stack monitor, which isokinetically samples the stack exhaust flow. Filter papers are periodically replaced in the monitor and isotopically analyzed for gamma emitting radionuclides to determine individual radionuclide total releases. Filter composites are periodically analyzed for pure beta emitting radionuclides (e.g. Sr-89/90).

Theoretically, Members of the Public could receive radiation doses from radioactivity released in gaseous effluents in three principal pathways, inhalation, ingestion and immersion. During normal plant operations, the major dose pathway is the immersion dose (external exposure) pathway. This was principally due to noble gas activity constituting over 99% of the radionuclides being released in gaseous effluents.

The second major dose pathway was from inhalation of small quantities of radionuclides released in gaseous effluents. The radionuclides which constituted the main fraction of dose via this pathway were the radioiodines (I-131, I-133) and Ba(La)-140. The last major dose pathway was from the infant ingestion of milk containing minute quantities of radionuclides, principally I-131 and Ba(La)-140, produced during plant operation. These pathways are illustrated in Figure 7.2.

After LACBWR was permanently shut down in April 1987, marked changes in the magnitude and composition of the radioactive gaseous effluents were observed. This is shown in Table 7-5 and Table 7-6. There are essentially no noble gas (Kr, Xe, Ar) nor radioiodine (I-131, 133) releases, since they are no longer being produced by the fission process, and any residual activity for these radionuclides (except Kr-85) within the stored spent fuel or plant systems has completely decayed to stable elements. In addition, many particulate-type radionuclide releases have diminished significantly due to radioactive decay and non-production. The Ba(La)-140 and Ru-103 have essentially completely decayed to stable elements. The particulate releases in gaseous

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[1]A technical specification submittal to remove the iodine monitoring requirement is planned since the iodines are no longer being produced and have decayed to stable elements.

effluents have decreased by a factor of about 60-100 times since shutdown. The tritium releases in gaseous effluents have significantly decreased. During the SAFSTOR period, the noble gas, radioiodine and tritium releases in gaseous effluents will not be evident under normal conditions. Although the particulate release activity may initially vary due to changes in plant tasks performed, it will continue to be lower than during plant operation, and the overall trend in releases activities and doses due to inhalation will continue to decrease during the long-term SAFSTOR period. The most important dose pathway for gaseous releases during SAFSTOR will be the inhalation pathway, although doses to Members of the Public from gaseous release would remain below 0.001 mRem per year, as illustrated in Table 7-7.

TABLE 7-5

TOTAL FISSION AND ACTIVATION PRODUCT ACTIVITY RELEASED IN GASEOUS EFFLUENTS
(Curies)

Radionuclide Category	1985	1986	Jan-Mar 1987	July-Sep 1987
Noble Gases	8580	3530	1793	N.D.
I-131 and I-133	1.1×10^{-2}	6.76×10^{-3}	1.24×10^{-3}	N.D.
H-3	34.8	12.1	8.71	N.D.
Particulates (T1/2 >8 days) including alpha (includes natural radionuclides)	2.63×10^{-3}	5.47×10^{-4}	2.97×10^{-4}	5.13×10^{-6}

N.D. = Not detectable above LLD.

TABLE 7-6

MAJOR RADIONUCLIDE BREAKDOWN
IN GASEOUS RELEASES
(Curies)

Radionuclide	1985	1986	Jan-Mar 1987	July-Sep 1987
Xe-135	5320	2110	988	N.D.
Xe-133	1120	428	252	N.D.
Kr-88	926	347	166	N.D.
Kr-85m	395	154	83.8	N.D.
Xe-131m	383	71.8	61.1	N.D.
Kr-87	373	156	83.6	N.D.
Xe-135m	152	50.6	38.7	N.D.
Xe-138	133	129	109	N.D.
Ar-41	60.4	44.2	6.1	N.D.
Xe-137	21.3	22.3	4.2	N.D.
Xe-133m	1.4E-1	13.4	1.8E-3	N.D.
I-133	6.3E-3	2.1E-3	5.1E-4	N.D.
I-131	4.9E-3	3.5E-3	7.2E-4	N.D.
Ba(La)-140	2.0E-3	3.0E-4	1.3E-4	N.D.
Co-60	2.3E-4	9.5E-5	1.3E-5	N.D.
Mn-54	8.2E-5	4.2E-5	6.1E-6	5.5E-7
Cs-137	5.8E-5	3.5E-5	4.3E-6	4.1E-7
Sr-89	4.7E-5	1.6E-5	1.3E-4	N.A.
Ce-144	3.2E-5	1.5E-6	5.3E-6	N.D.
Co-58	3.2E-5	1.2E-5	1.8E-6	N.D.
Cr-51	1.1E-5	6.1E-6	4.6E-6	N.D.
Ru(Rh)-106	4.3E-5	1.1E-5	N.D.	N.D.
Ce-141	7.8E-6	2.3E-6	1.5E-6	N.D.
Ru-103	3.4E-6	4.2E-6	5.7E-7	N.D.
Sr-90	7.5E-7	N.D.	1.6E-7	N.A.
Alpha (including naturals)	1.2E-4	2.1E-5	5.1E-6	4.1E-6
H-3	34.8	12.1	8.7	N.D.

N.D. = None detectable above LLD.

N.A. = Not available.

TABLE 7-7

CALCULATED DOSES TO MEMBERS OF THE PUBLIC
FROM GASEOUS EFFLUENT RELEASES
(mRem)

	1985	1986	Jan-Mar 1987	July-Sep 1987
Whole Body (Noble Gases)	1.232	0.662	0.166	0.000
Skin (Noble Gases)	1.176	0.632	0.158	0.000
Highest Organ (I-131, I-133 and/or H-3 & Particulates)	0.563 (Child Thyroid)	0.470 (Child Thyroid)	0.083 (Child Thyroid)	0.001 (Child Bone)

7.4 Solid Radioactive Waste Processing and Shipments

During the SAFSTOR period, the principal types of radioactive solid waste which will be processed and shipped to a suitable disposal facility will be low level radioactive wastes principally with radioactivity content less than Class C (10 CFR 61) wastes. This will include:

1) Dry Active Wastes (DAW), normally Class A unstable,

2) Dewatered Spent Demineralizer resins and filtration media, normally Class A, B or C stable, and

3) Contaminated or irradiated plant system components, normally Class B or C, stable.

All solid radioactive wastes will be processed and shipped in accordance to an approved Process Control Program. The average volumes of low level solid radioactive waste shipped during the period 1977-1986 and 1987, when LACBWR was operating, are listed on Table 7-8. Projected low level solid waste shipments for the SAFSTOR period are listed on Table 7-9. Although the waste volumes and activities should initially vary, they will be below historical operating average annual volumes and activities. This will be principally because of reduced DAW generation, reduced demineralizer resin and filtration component usage, and because of radioactive decay of Co-60, Fe-55, Mn-54, Ce-141/144, and other radionuclides. During the longer term SAFSTOR period, the average waste volumes generated may be approximately 7.5 m^3/yr, which

would be a factor of about 3.5 less than the historical operating annual average. The average annual activity in waste should be a factor of up to 6 times less than in waste shipped during periods of operation.

It is estimated that the average annual dose to any Member of the Public, due to direct radiation exposure from waste processing and shipment activities during the SAFSTOR period, will not exceed 1 mRem/year.

TABLE 7-8

LOW LEVEL RADIOACTIVE WASTE SHIPMENTS FROM LACBWR DURING OPERATION

	1977-1986	1987
Volume/yr (m^3)	25.22 ± 21.53	29.31
Activity/yr (Ci)	158.78 ± 162.33	95.0

TABLE 7-9

PROJECTED LOW LEVEL RADIOACTIVE WASTE SHIPMENTS FROM LACBWR DURING SAFSTOR

	1988	1989	1990	1990-2010
Volume/yr (m^3)	10	21	5	5 - 7.5(1)
Activity/yr (Ci)	140	65	40	15 - 20

(1) Assumes periodic shipments of contaminated and/or irradiated components no longer required for SAFSTOR maintenance and surveillance.

7.5 Offsite Dose Calculation Manual (ODCM)

The purpose of the Offsite Dose Calculation Manual (ODCM) is to describe the methodology and parameters to be used in the calculation of instantaneous release rate monitor alarm setpoints and the dose commitments to Members of the Public beyond the Effluent Release Boundary due to the release of radioactive materials in liquid and gaseous effluents from LACBWR. The ODCM contains calculational methods for determining periodically the effluent radiation monitors' alarm setpoints, and also provides site specific dose factors for the LACBWR site. The site specific dose factors enable the calculation of cumulative dose contributions to Members of the Public on a quarterly and annual basis for principal pathways of exposure, as required by Technical Specifications, to demonstrate compliance with the requirements listed in Section 7.1. The ODCM also contains a description of the Radiological Environmental Monitoring (REM) Program.

7.6 Estimated Doses to Members of the Public During SAFSTOR

The estimated average dose to any Member of the Public for the initial few years of the SAFSTOR period is expected to be less than 2 mRem/year. For the long term SAFSTOR period, the estimated average dose is expected to decrease well below 1 mRem/year, due to nonproduction and continued radioactive decay of predominant residual radionuclides (e.g., Co-60, Fe-55, Ag-110m, Mn-54, Cs-137). The initial estimated average annual dose is summarized as follows:

Direct External Radiation Dose (Primarily from LLRW shipments)	1.000 mRem/yr
Liquid Releases (Fish - Adult Ingestion Pathway)	0.500 mRem/yr
Liquid Releases (Immersion or External Dose)	0.005 mRem/yr
Liquid Releases (Ingestion)	0.001 mRem/yr
Gaseous Releases (Inhalation)	0.001 mRem/yr
Gaseous Releases (Ingestion)	0.000 mRem/yr
Gaseous Releases (Immersion or External Dose)	0.000 mRem/yr
TOTAL	1.507 mRem/yr

7.7 Nonradioactive Wastes

7.7.1 Chemical Wastes

Chemical wastes which could be produced during the SAFSTOR period will be processed and handled according to appropriate hazards classification, and discharged to the environment in accordance with the requirements of the Wisconsin Pollution Discharge Elimination System (WPDES) Permit. The discharge to the Mississippi River comes from one source--the plant circulating cooling water. No chlorine or other chemicals are used to treat cooling water, and therefore none are discharged. Decontamination chemicals, laboratory chemicals (reagents), and minute quantities of residual Mg CrC_4, $K_2Cr_2O_7$ and Na_2 CrO_4 which have been used in selected plant systems for corrosion inhibition are periodically discharged with the liquid radioactive waste batch discharges. Morpholine, which is added to the heating boiler during boiler usage, may evaporate to the atmosphere. Sulfuric Acid and Sodium Hydroxide used for makeup demineralizer regenerations are not discharged to the river, but are neutralized in a settling pond. Waste solvents, paints and degreasers will be disposed of in hazardous material collection containers. Any chemicals used for system decontamination during SAFSTOR will be processed using appropriate demineralizers or filtration systems which will be processed as solid wastes.

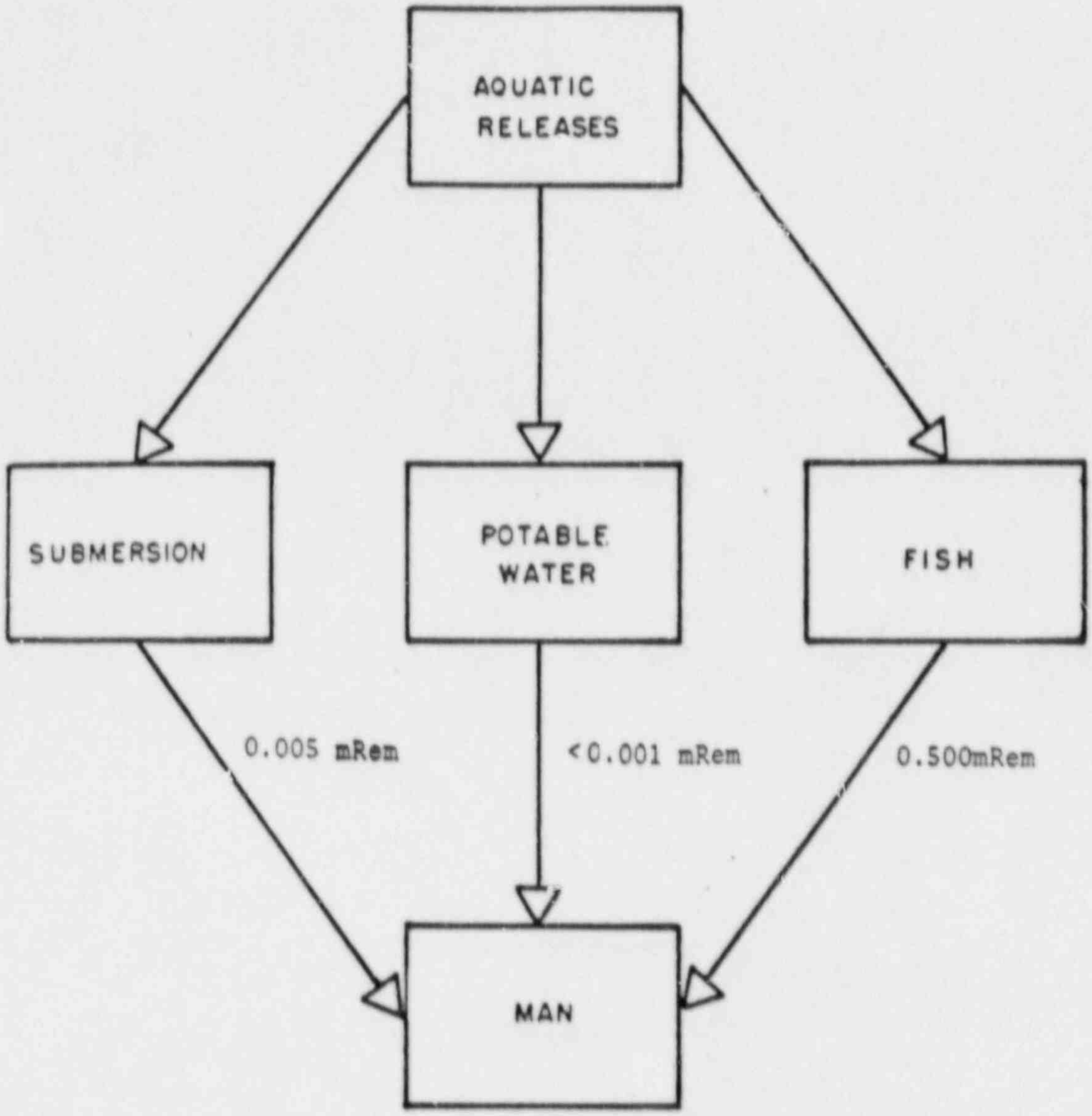
No sanitary wastes are discharged to the river. All of the waste from toilets and sinks (except for decontamination sinks) are discharged to underground septic tanks located within the Owner Controlled Area. Plant shower discharges (except for decontamination showers) are discharged into the septic system. Decontamination sinks and showers discharge to the liquid radioactive waste system.

7.7.2 Noncombustible Solid Wastes

The disposal of noncombustible solid wastes has changed from that described in the original Environmental Report. Nonradioactive noncombustible solid wastes are now deposited at a landfill in Genoa, instead of at the Genoa Unit No. 3 ash disposal area, with the exception of metal wastes, which are collected by a scrap metal dealer. Any asbestos will be disposed of in accordance with the Dairyland Power Cooperative asbestos control program.

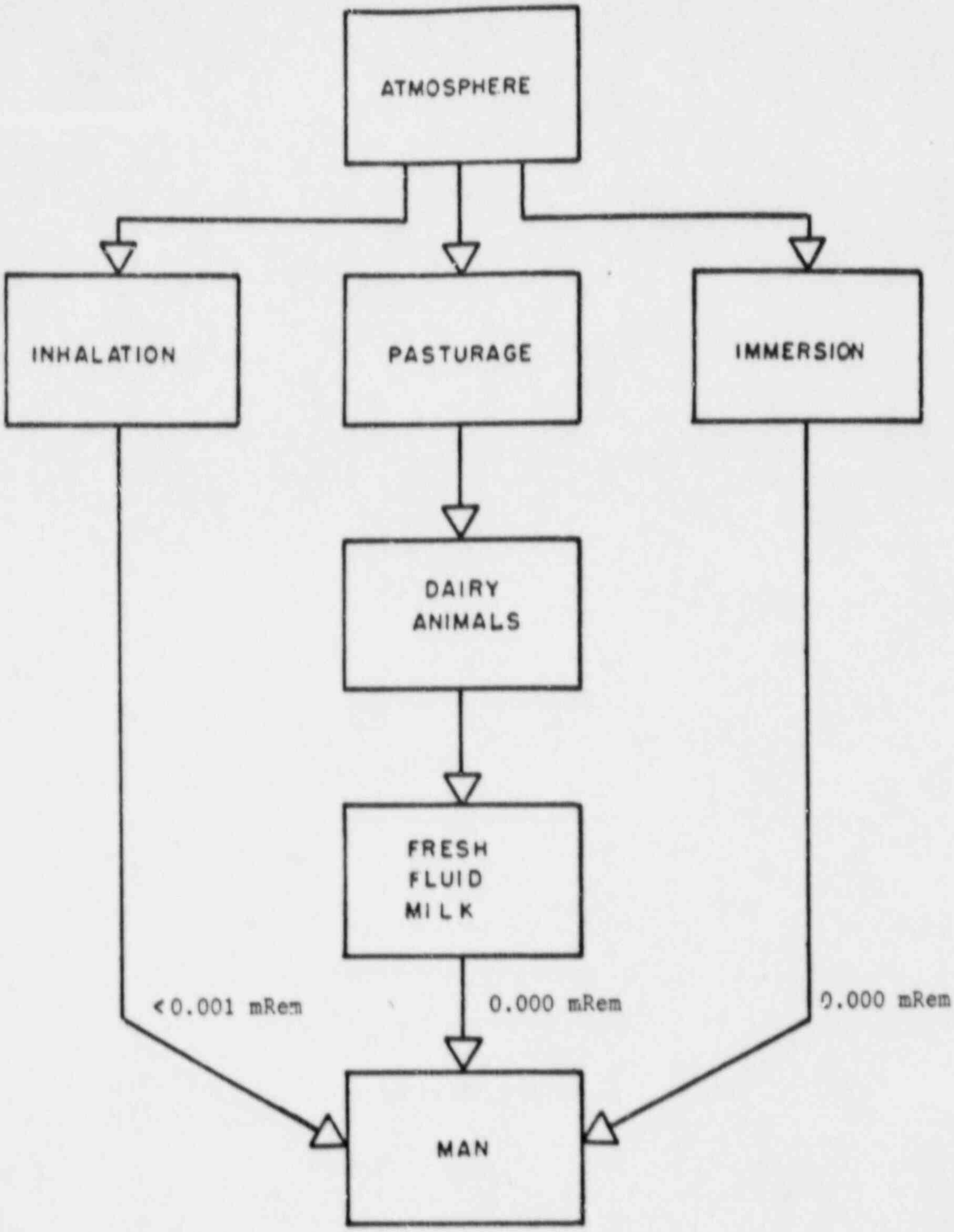
7.8 Comparison of Expected SAFSTOR Releases to Releases During Plant Operation

The release of radioactive material to the environment during the SAFSTOR period will be significantly less than during plant operation. There will be no noble gas nor radioiodine releases in either liquid or gaseous effluents during SAFSTOR. The shorter-lived particulate-type radionuclides such as Ba(La)-140, Zr(Nb)-95, Sr-89, Ru-103, Ce-141, Fe-59 and Cr-51 will not be available for release to the environment about 6 months to 1 year after shutdown. The intermediate-lived, particulate-type radionuclide sources such as Ag-110m, Ce-144, Mn-54, Ru-106 and Cs-134 will decrease by factors ranging from 29 to several thousand within ten years after shutdown. Cobalt-60 and Iron-55 which are predominant radionuclides in liquid waste discharges and solid wastes will decrease by a factor of 3.7 and 13 respectively, within ten years after shutdown, due to radioactive decay.



Potential Radiation Exposure Pathways
and Average Annual Doses to Members of the Public
for Liquid Releases from LACBWR
During SAFSTOR

FIGURE 7.1



Potential Radiation Exposure Pathways
and Average Annual Doses to Members of the Public
for Gaseous Releases from LACBWR
During SAFSTOR

FIGURE 7.2

8.0 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

8.1 A Radiological Environmental Monitoring (REM) Program will be conducted to comply with Technical Specifications and 10 CFR 50 Appendix I during the SAFSTOR period. Environmental samples will be taken within the surrounding areas of the plant. These samples will be analyzed to determine any effects plant effluent releases may have on the environment and the public. These samples will include (1) direct radiation measurement devices, (2) air particulate samples, (3) river water and sediment samples, (4) fish samples, (5) vegetation or foliage samples, taken on a periodic basis as specified by Technical Specifications.

8.2 The results of the REM program sample analyses for the period July-September 1987 are listed on Table 8-1. Most of the results demonstrate that no plant attributable radioactivity nor radiation is detectable above the normal background radiation levels. Direct radiation levels in the environment, as measured by TLD's, ranged from 14.8 ± 3.0 mRem to 26.0 ± 9.5 mRem/quarter. The average direct radiation measurement was 19.8 ± 3.0 mRem/quarter. The beta particulate activities in environmental air samples remained at approximately 0.02 pCi/m^3 on the average, with no significant difference between the La Crosse air sample activities, nor samples obtained closer to the site. No airborne I-131 activity above the MDA was detected in any charcoal cartridge samples. No detectable gamma emitting radionuclide activity was found in air particulate composite samples. No detectable gamma emitting radionuclide activity was found in fish samples nor milk samples. No significant activity was detected in river water samples.

Mississippi River bottom sediment samples exhibited a small amount of plant attributed radionuclide accumulation, principally at the outfall of the circulating water system. The radionuclide adsorption into sediments is very localized and does not exhibit an increase over previous years' data. This will continue to be monitored and is expected to decrease during the SAFSTOR period.

These results are similar to results obtained in environmental sample analyses during plant operation.

These results should be typical of results of environmental sample results for the entire SAFSTOR period. In other words, no detectable plant attributed radioactivity should be present in environmental samples obtained during SAFSTOR.

TABLE 8-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM
RESULTS SUMMARY

(July-September 1987)

Sample Type	Range of Sample Results	Average Sample Result
Direct Radiation (TLD's)	14.8 ± 3.0 to 26.0 ± 9.5 mRem	19.8 ± 3.0 mRem
Air Particulate Filter (Gross Beta)	0.009 ± 0.002 to 0.050 ± 0.004 pCi/m³	0.023 pCi/m³
Air Particulate Filter (Gamma isotopic)	<3.66 E-4 to 4.43 E-3 pCi/m³	<7.00 E-4 pCi/m³
Air Charcoal Cartridge (I-131)	<1.86 E-3 to <4.11 E-3 pCi/m³	<3.00 E-3 pCi/m³
Milk Sample (Gamma Isotopic)	<2.25 to <25.4 pCi/l	<5.5 pCi/l
River Water (Gamma Isotopic)	<3.65 to <24.14 pCi/l	<10 pCi/l
Fish - Edible portion (Gamma isotopic)	<6.56 to <24.81 pCi/kg	<12.5 pCi/kg
River Sediments - Outfall + River Proper (Gamma isotopic)	5.90 E1 to 6.81 E3 pCi/kg	2.32 E3 ± 3.89 E3 pCi/kg

9.0 POTENTIAL ACCIDENTS DURING SAFSTOR

9.1 Introduction

The probability of an accident occurring during the SAFSTOR period is considerably less than during plant operation. The focus of the potential accidents has also changed. During operation, the focus was on minimizing the plant transient and cooling the reactor core. During SAFSTOR, the only major concern is protecting the fuel in the Fuel Element Storage Well.

The fuel in the well, while not benign, is not as much a hazard as the fuel in the operating reactor was. Since April 30, 1987, the fission product inventory has decreased and the decay heat generation is significantly less. These factors reduce the consequences of any accident affecting the fuel. As time passes, the consequences will continue to decrease.

The reactor's design basis accidents were reviewed to determine which could still occur during SAFSTOR. Some other accident scenarios which were not previously considered design basis accidents were also evaluated. A list of 8 postulated accidents was identified. These events are:

- Spent Fuel Handling Accident
- Shipping Cask or Heavy Load Drop into FESW
- Loss of FESW Cooling
- FESW System Pipe Break
- Uncontrolled Liquid Waste Discharge
- Loss of Offsite Power
- Earthquakes
- Wind and Tornado

Each of these postulated events was evaluated based on the revised plant status to identify their potential consequences during the SAFSTOR period. The following sections discuss these accidents.

One additional event was examined - a fire. The potential safety consequences of any fire fall within the scope of other evaluated events.

9.2 Spent Fuel Handling Accident

This accident postulates a fuel assembly falling from the hoist into the Fuel Element Storage Well. The probability of this accident is extremely small, since minimal fuel handling will be performed during the SAFSTOR period until the fuel assemblies are removed from the FESW. Periodic inspections may be conducted during the years the fuel remains onsite. In the almost 20 years of operation and associated fuel handling at LACBWR, no fuel assemblies were ever dropped.

In this event, it is assumed that the cladding of all the pins in 2 fuel assemblies ruptures. The fuel handling crew evacuates when the local area radiation monitor alarms. Containment Building ventilation would isolate on high activity, but for this analysis, no containment integrity is assumed.

The assumptions used in evaluating this event during SAFSTOR were similar to those used in the FESW reracking analyses.[1 2] The fuel inventory calculated for October 1987 was used. The only significant gaseous fission product available for release is Kr-85. The plenum or gap Kr-85 represents about 15% (215.7 Curies) of the total Kr-85 in the fuel assemblies. However, for conservatism and commensurate with Reference 1, 30% of the total Kr-85 activity, or 431.4 Curies, is assumed to be released in this accident scenario.

No credit was taken for decontamination in the FESW water or for containment integrity, so all the activity was assumed to be released into the environment. Meteorologically stable conditions at the Exclusion Area Boundary (1109 ft, 338m) were assumed, with a release duration of 2 hours commensurate with 10 CFR 100 and Regulatory Guides 1.24 and 1.25.

A stack release would be the most probable, but a ground release is not impossible given certain conditions. Therefore, offsite doses were calculated for 3 cases. The first is at the worst receptor location for an elevated release, which is 500m E of the Containment Building. The next case is the dose due to a ground level release at the Exclusion Area Boundary. The maximum offsite dose at the new proposed Emergency Planning Zone boundary[3] for a ground level release is also calculated. Adverse meteorology is assumed for all cases.

Elevated Release

Average Kr-85 Release Rate

$$\frac{431.4 \text{ Curies}}{2 \text{ hrs.} \times 3600 \text{ sec/hr}} = 6.00 \text{ E-2 Ci/sec}$$

Worst Case $\frac{X}{Q}$ for 0-2 hours at 500m E = 2.3 E-4 sec/m^3

Kr-85 average concentration at 500m E

$$6.00 \text{ E-2 Ci/sec} \times 2.3 \text{ E-4 sec/m}^3 = 1.38 \text{ E-5 Ci/m}^3$$

Immersion Dose Conversion at 500m E

Kr-85 Gamma Whole Body Dose Factor (Regulatory Guide 1.109)

$$1.61 \text{ E+1} \frac{\text{mRem/yr}}{\mu\text{Ci/m}^3} \times 10^6 \frac{\mu\text{Ci}}{\text{Ci}} \times 1.142 \text{ E-4} \frac{\text{yr}}{\text{hr}} = 1{,}839 \frac{\text{mRem/hr}}{\text{Ci/m}^3}$$

Whole Body Dose at 500m E

$$1839 \frac{\text{mRem/hr}}{\text{Ci/m}^3} \times 1.38 \text{ E-5 Ci/m}^3 \times 2 \text{ hr} = 0.05 \text{ mRem}$$

Kr-85 Beta/Gamma Skin Dose Factor (Regulatory Guide 1.109)

$$1.34\ E{+}3\ \frac{mRem/yr}{\mu Ci/m^3} \times \frac{10^6\ \mu Ci}{Ci} \times 1.142\ E{-}4\ \frac{yr}{hr} = 1.53\ E5\ \frac{mRem/hr}{Ci/m^3}$$

Skin Dose at 500m E

$$1.53\ E5\ \frac{mRem/hr}{Ci/m^3} \times 1.38\ E{-}5\ Ci/m^3 \times 2\ hr = 4.2\ mRem$$

Ground Level Release at EAB

Worst Case $\frac{X}{Q}$ for 2 hrs at 338m NE or 338m SSE, using Regulatory Guide 1.25

$2.2\ E{-}3\ \frac{sec}{m^3}$

Whole Body Dose at 338m

0.49 mRem

Skin Dose at 338m

40.4 mRem

Ground Level Release at Proposed Emergency Planning Zone Boundary

Worst Case $\frac{X}{Q}$ for 2 hrs at 100m E

$1.02\ E{-}2\ \frac{sec}{m^3}$

Whole Body Dose at 100m E

2.25 mRem

Skin Dose at 100m E

187 mRem

As can be seen, the estimated maximum whole body dose is more than a factor of 11,000 below the 10 CFR 100 dose limit of 25 Rem (25,000 mRem) to the whole body within a 2-hour period.

9.3 Shipping Cask or Heavy Load Drop into FESW

This accident postulates a shipping cask or other heavy load falling into the Fuel Element Storage Well. Reference 1 stated that extensive local rack deformation and fuel damage would occur during a cask drop accident, but with an additional plate (that was installed during the reracking) in place, a dropped cask would not damage the pool liner or floor sufficiently to adversely affect the leak-tight integrity of the storage well (i.e., would not cause excessive water leakage from the FESW).

For this accident, it is postulated that all 333 spent fuel assemblies located in the FESW are damaged. The cladding of all the fuel pins ruptures. The same assumptions used in the Spent Fuel Handling Accident (Section 9.2) are used here. A total of 35,760 Curies of Kr-85 is released within the 2-hour period. The doses calculated are as follows:

Elevated Release

Whole Body Dose at 500m E

4.2 mRem

Skin Dose at 500m E

350 mRem

Ground Level Release at EAB

Whole Body Dose at 338m

40.2 mRem

Skin Dose at 338m

3.34 Rem

Ground Level Release at Proposed Emergency Planning Zone Boundary

Whole Body Dose at 100m E

186 mRem

Skin Dose at 100m E

15.6 Rem

As can be seen, the estimated offsite doses for the cask drop accident are below the 10 CFR 100 limits. The postulated maximum whole body dose is more than a factor of 100 below the 10 CFR 100 limit of 25 Rem (25,000 mRem).

9.4 Loss of FESW Cooling

This accident postulates a loss of FESW cooling. The most likely causes of a loss of cooling are:

1) Both FESW pumps fail or FESW piping has to be isolated for maintenance;

2) The Component Cooling Water (CCW) System is out of service due to failure of both pumps or other reason. The CCW System removes heat from the FESW cooler.

3) The Low Pressure Service Water (LPSW) System is out of service due to failure of both pumps or other reason. The LPSW System removes heat from the CCW coolers.

If the third possibility is the cause, cooling to the CCW coolers can be restored by cross-connecting the High Pressure Service Water System to the coolers, in lieu of LPSW.

A calculation was performed to determine FESW heatup rate if active cooling were lost. The decay heat generation rate for January 1, 1988, was used. Pool boiling would commence approximately 5 days after the loss of cooling. The tops of the control rods stored in the fuel racks would become uncovered after 13.6 days. After 23.1 days, the top of the fuel would be exposed. If water is added to the FESW at any time during this period, these consequences would be delayed.

Substantial time is therefore available for restoration of FESW cooling. No immediate action is necessary during this postulated accident.

9.5 FESW Pipe Break

This accident postulates a break in the FESW system piping, other than in the pump discharge piping between the redundant check valves and the pool liner. A load analysis was performed on this approximately 20 feet of piping. It was concluded that all stresses are within ASME Code allowable. (Reference 1 calls this line the spent fuel pool drain line.) The series check valves were added during the 1980 FESW reracking.

If the postulated break occurs, the lowest the FESW could drain is approximately 679'. At this level all spent fuel will remain covered. The control rods which are currently stored in the fuel racks will be partially uncovered. The tops of the control rods are about elevation 686'.

The operator would be alerted to this accident by receipt of the FESW Level Lo/High alarm. Any makeup water added may run out the break, depending on the size of the break.

A calculation has been performed to determine the radiation levels due to the exposed control rods. In the vicinity of most of the FESW piping and isolation valves, the radiation dose would not be substantially increased due to the loss of water.

A repair team should be able to access the break location or piping isolation valves and either isolate the break or effect temporary repairs. FESW level could then be restored to normal.

There would be no immediate urgency to restore the level. The partially uncovered control rods only create a local problem. No offsite release is associated with this event. Active FESW cooling would be lost during this accident, but as discussed in Section 9.4, considerable time is available to take action. Due to the lesser water volume to act as the heat sink and reduced fuel coverage, less time would be available to restore cooling during this accident scenario than in just a loss of FESW cooling event, but boiling would not commence for more than 1 day. As with the loss of FESW cooling event, if water is added to the FESW, any consequences of water heatup can be delayed or prevented. Water can be added from the Demineralized Water System or the Overhead Storage Tank.

9.6 Uncontrolled Waste Water Discharge

This accident postulates that an operator starts pumping an unsampled or incorrectly analyzed Waste Water or Retention Tank to the river. If the contents of the tank are of normal activity, this event will not be detected until the lineup is being secured after pumping, if then.

If the liquid in the tank is of high activity, the waste water monitor will alarm and the Auto Flow Control Valve will automatically close, terminating the discharge. The Turbine Condenser Cooling Water Monitor will also alarm, if the activity is high enough. If the automatic valve does not close, an operator will try to close it from the Control Room. If it cannot be closed, an operator will close a local valve or secure the pump to terminate the discharge.

After the discharge is terminated, a sample of the tank will be taken to analyze the radioactivity concentrations and effective MPC_w for the uncontrolled release. Waste water is diluted by LACBWR Circulating Water and Low Pressure Service Water flow, in addition to circulating water from the adjacent coal-fired plant, prior to being discharged into the river.

The environmental consequence of this postulated event would be less severe than that discussed in the original Environmental Report. The probability of an event resulting in an actual discharge has been reduced by the installation of the Auto Flow Control Valve.

9.7 Loss of Offsite Power

This accident postulates a loss of offsite power. If both Emergency Diesel Generators and a High Pressure Service Water (HPSW) Diesel start, adequate FESW cooling can be provided and adequate instrumentation is available to monitor FESW conditions from the Control Room. All that is needed is for an operator to cross-connect HPSW to the Component Cooling Water (CCW) coolers.

If an HPSW Diesel and 1B Emergency Diesel Generator start, FESW cooling can be provided. If 1A Emergency Diesel Generator (EDG) starts, but 1B does not, adequate cooling can be provided only if the essential buses are tied together.

If one or more EDG's start, but neither HPSW diesel starts, no ultimate heat sink for the FESW would be available. The consequences would be the same as in the Loss of FESW Cooling Event (Section 9.4).

If neither EDG can be started, neither FESW or CCW pump can run. The consequences again are the same as a Loss of FESW Cooling Event, with the additional complication that some instrumentation will be lost immediately and others after the station batteries are depleted. The operator would have to check the FESW locally periodically.

As discussed in Section 9.4, the FESW would not start boiling for about 5 days, the tops of the control rods would become uncovered in 13.6 days and the tops of the fuel would be exposed in 23.1 days. Therefore, no immediate action needs to be taken and sufficient time is available to take corrective actions to restore power.

9.8 Seismic Event

This accident postulates that a design basis earthquake occurs. The magnitude of the seismic event and damage incurred is the same as that assumed during the Systematic Evaluation Program (SEP) and the Consequence Study prepared as part of the SEP Integrated Assessment (References 4-7). The major concern of the previous evaluation was to safely shut down the plant and maintain adequate core cooling to prevent fuel damage. The focus now is to prevent damage to the fuel stored in the Fuel Element Storage Well.

Seismic analysis has shown the Containment Building structure, LACBWR stack and Genoa Unit 3 stack are capable of withstanding the worst postulated seismic event at the LACBWR site. Reference 1 documented that the storage well, itself, the racks and the bottom-entry line between the check valves and the storage well can withstand the postulated loads.

The potential consequences of most interest due to a seismic event could include loss of all offsite and onsite power and a break in the FESW System piping. This event, therefore, can be considered as a combination of a Loss of Power Event (Section 9.7) and FESW Line Break (Section 9.5). As with these individual events, considerable time is available for response to a seismic event, with the FESW System pipe break requiring the earlier response. Access to the break location may be more difficult following a seismic event due to failure of other equipment in the plant. The time available, though, should be more than sufficient to initiate mitigating actions. (Refer to Section 9.5).

9.9 Wind and Tornado

This accident postulates that design basis high wind or tornado event occurs. The magnitude of the event and damage incurred is the same as that assumed during the Systematic Evaluation Program (SEP) and the Consequence Study prepared as part of the SEP Integrated Assessment (References 4-9). The major concern of the previous analyses was to ensure that adequate cooling of the reactor core was maintained. The focus now is to prevent damage to the fuel stored in the Fuel Element Storage Well.

The previous evaluations determined that the Containment Building would withstand this event. The Turbine Building, Diesel Building, Cribhouse and switchyard may be damaged. The probability of the LACBWR or Genoa Unit 3 stacks failing and impacting the Containment Building was determined to be low enough that it need not be considered. Personnel outside the Containment Building may not survive.

The potential plant consequence of primary concern is the loss of all offsite and onsite power. As discussed in Section 9.7, Loss of Offsite Power, considerable time is available before action must be taken to protect the fuel.

9.10 References

1) NRC Letter, Ziemann to Linder, dated February 4, 1980.

2) NRC Letter, Reid to Madgett, dated October 22, 1975.

3) DPC Letter, Taylor to Document Control Desk, LAC-12377, dated September 29, 1987.

4) DPC Letter, Linder to Paulson, LAC-10251, dated October 11, 1984.

5) NRC Letter, Zwolinski to Linder, dated January 16, 1985.

6) DPC Letter, Linder to Zwolinski, LAC-10639, dated March 15, 1985.

7) NRC Letter, Zwolinski to Taylor, dated September 9, 1986.

8) DPC Letter, Taylor to Zwolinski, LAC-12052, dated January 14, 1987.

9) NRC Letter, Bernero to Taylor, dated April 6, 1987.

10.0 REGULATORY GUIDE 1.86

Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors," issued June 1974 currently provides the only NRC-approved guidance for license termination and acceptable releasable surface contamination levels. The NRC proposed rule on Decommissioning Criteria for Nuclear Facilities redefines the decommissioning options listed in Regulatory Guide 1.86 and discusses the submittals required for plant decommissioning. It does not, however, propose revised acceptable surface contamination levels.

Any material released from LACBWR for unrestricted use will therefore meet the established criterion of Table 1 of Regulatory Guide 1.86, which is reproduced here for reference.

ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDE [a]	AVERAGE [b,c]	MAXIMUM [b,d]	REMOVABLE [b,e]
U-nat, U-235, U-238, and associated decay products	5,000 dpm α/ 100 cm^2	15,000 dpm α/ 100 cm^2	1,000 dpm α/ 100 cm^2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1000 dpm/ 100 cm^2	3000 dpm/ 100 cm^2	200 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5000 dpm β-γ/ 100 cm^2	15,000 dpm β-γ/ 100 cm^2	1000 dpm β-γ/ 100 cm^2

[a] Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

[b] As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c] Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[d] The maximum contamination level applies to an area of not more than 100 cm^2.

[e] The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

LA CROSSE BOILING WATER REACTOR
(LACBWR)

DECOMMISSIONING PLAN

December 1987

Prepared by the LACBWR Staff

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1. INTRODUCTION

The Decommissioning Plan describes Dairyland Power Cooperative's (DPC) plans for the future disposition of the La Crosse Boiling Water Reactor (LACBWR). DPC intends to place LACBWR in the SAFSTOR mode, so this plan describes the plant's status and provides a safety analysis for the SAFSTOR period. A separate preliminary DECON Plan is being submitted to outline Dairyland Power Cooperative's intention to ultimately decommission the plant and site to radiologically releasable levels and terminate the license in accordance with Nuclear Regulatory Commission (NRC) requirements.

This Decommissioning Plan concentrates on the status of LACBWR while the reactor fuel remains in the Fuel Element Storage Well. There are 333 activated fuel assemblies onsite. The plan at this time is to store the fuel in the existing Fuel Element Storage Well. DPC currently expects the fuel to remain onsite until a federal repository is established and ready to receive LACBWR fuel.

1.1 SELECTION OF SAFSTOR

The Nuclear Regulatory Commission (NRC) proposed rule on Decommissioning Criteria for Nuclear Facilities identifies 3 major classifications of decommissioning alternatives. They are DECON, SAFSTOR, and ENTOMB. The proposed rule defines the alternatives as follows:

> DECON is the alternative in which the equipment, structures, and portions of a facility and site containing radioactive contaminants are removed or decontaminated to a level that permits the property to be released for unrestricted use shortly after cessation of operations.
>
> SAFSTOR is the alternative in which the nuclear facility is placed and maintained in such condition that the nuclear facility can be safely stored and subsequently decontaminated (deferred decontamination) to levels that permit release for unrestricted use.
>
> ENTOMB is the alternative in which radioactive contaminants are encased in a structurally long-lived material, such as concrete. The entombed structure is appropriately maintained and continued surveillance is carried out until the radioactivity decays to a level permitting unrestricted release of the property. This alternative would be allowable for nuclear facilities contaminated with relatively short-lived radionuclides such that all contaminants would decay to levels permissible for unrestricted use within a period on the order of 100 years.

For a power reactor the choice is either DECON or SAFSTOR. Due to some of the long-lived isotopes in the reactor vessel and internals, ENTOMB, by itself, is not an allowable alternative under the proposed rule.

The choice between SAFSTOR and DECON must be made based on a variety of factors including availability of fuel and waste disposal, land use, radiation exposure, waste volumes, economics, safety, and availability of experienced personnel. Each alternative has advantages and disadvantages. The best option for a specific plant has to be chosen based on an evaluation of the factors involved.

1. INTRODUCTION - (cont'd)

The overriding factor affecting the decommissioning decision for LACBWR is that a federal repository is not expected to be available for fuel storage for about 20 years. With the fuel in the Fuel Element Storage Well, the only possible decommissioning option is SAFSTOR. Only limited decontamination and dismantling of unused systems can be performed during this period.

There are other reasons to choose the SAFSTOR alternative. The majority of piping radioactive contamination is Co-60 (5.27 yr half-life) and Fe-55 (2.7 yr half- life). If the plant is placed in SAFSTOR for 50 years, essentially all the Co-60 and Fe-55 will have decayed to stable elements. Less waste volume will be generated and radiation doses to personnel performing the decontamination and dismantling activities will be significantly lower. Therefore, delayed dismantling supports the ALARA (As Low As Reasonably Achievable) goal. The reduction in dismantling dose exceeds the dose the monitoring crew receives during the SAFSTOR period.

The decommissioning cost estimate is discussed in Section 6.7. The majority of studies show that while the total cost of SAFSTOR with delayed DECON is greater than immediate DECON, the present value is less for the SAFSTOR with delayed DECON option.

The main disadvantage of delayed DECON is that the plant continues to occupy the land during the SAFSTOR period. The land cannot be released for other purposes. DPC also operates a 350 MWe coal-fired power plant on the site. Due to the presence of the coal-fired facility, DPC will continue to occupy and control the site, regardless of the nuclear plant's status. Therefore, the continued commitment of the land to LACBWR during the SAFSTOR period is not a significant disadvantage.

A second disadvantage of delaying the final decommissioning is that the people who operated the plant would not be available for the DECON period. When immediate DECON is selected, some of the experienced plant staff would be available for the dismantling. Their knowledge of plant characteristics and events could be extremely helpful. In the absence of these knowledgeable people, all information has to be obtained from plant records. When SAFSTOR is chosen, efforts must be made to maintain excellent records to compensate for the lack of staff continuity.

The remaining factor to be discussed is safety. As of August 1987, 43 power reactors have been shut down worldwide, 19 of which are in the United States. All 3 methods of decommissioning are being used. Experience has shown that all can be used safely.

The Nuclear Regulatory Commission issued its Waste Confidence Decision in the Federal Register on August 31, 1984. In it, the NRC found "reasonable assurance that, if necessary, spent fuel generated in any reactor can be stored safely and without significant environmental impacts for at least

30 years beyond the expiration of that reactor's operating license at that reactor's spent fuel storage basin, or at either onsite or offsite independent spent fuel storage installations." Therefore, DPC's plan to maintain the activated fuel at LACBWR, until a federal repository is ready to accept the fuel, is acceptable from the safety standpoint, as well as necessary from the practical standpoint.

A possible fourth decommissioning alternative exists which combines some features of SAFSTOR and DECON. The possibility exists to use the secondary side of the plant with a new fossil-fired steam supply system. While DPC is not planning on pursuing this option at this time, it should not be eliminated as an alternative.

After evaluating the factors involved in selecting a decommissioning alternative, Dairyland Power Cooperative decided to choose an approximately 30-50 year SAFSTOR period, followed by DECON. The exact duration of the SAFSTOR period will be dependent on the availability of the federal fuel repository, availability of waste disposal, economics, personnel exposure, and various institutional factors. If any major changes are made in DPC's decommissioning plans, a revision to this plan will be prepared.

1.2 REFERENCES

1) Nuclear Regulatory Commission, proposed rule on Decommissioning Criteria for Nuclear Facilities, Federal Register, Vol. 50, No. 28, February 11, 1985.

2) Nuclear Regulatory Commission, Waste Confidence Decision, Federal Register, Vol. 49, No. 171, August 31, 1984.

3) "Decommissioning - Demonstrating the Solution to a Problem for the Next Century," Nuclear Engineering International, Vol. 32, No. 399, October 1987, p. 48.

4) Proceedings from the 1987 International Decommissioning Symposium, Conf-871018, October 4-8, 1987.

2. LA CROSSE BOILING WATER REACTOR OPERATING HISTORY

2.1 INTRODUCTION

The La Crosse Boiling Water Reactor (LACBWR) is owned and was operated by Dairyland Power Cooperative.

The LACBWR was a nuclear power plant of nominal 50 Mw electrical output, which utilized a forced-circulation, direct-cycle boiling-water reactor as its heat source. The plant is located on the east bank of the Mississippi River in Vernon County, Wisconsin, approximately 1 mile south of the village of Genoa, Wisconsin, and approximately 19 miles south of the city of La Crosse, Wisconsin.

The plant was one of a series of demonstration plants funded in part by the U.S. Atomic Energy Commission (AEC). The nuclear steam supply system and its auxiliaries were funded by the AEC, and the balance of the plant was funded by the Dairyland Power Cooperative. The Allis-Chalmers Company was the original licensee; the AEC later sold the plant to the Dairyland Power Cooperative (DPC) and provided them with a provisional operating license.

2.2 INITIAL CONSTRUCTION AND LICENSING HISTORY

Allis-Chalmers, under a contract with the Atomic Energy Commission, had the responsibility for the design, fabrication, construction, and startup of the reactor. Allis-Chalmers retained Sargent & Lundy Engineers as architect-engineers for the project and the Maxon Construction Company as constructors. Dairyland Power Cooperative of La Crosse, Wisconsin, furnished the plant site and all equipment, facilities, and services necessary for a complete and operable nuclear plant.

Allis-Chalmers Atomic Energy Division and the United States Atomic Energy Commission (USAEC) entered into a contract, AT(11-1)-850, on June 6, 1962, to construct a second round demonstration nuclear power plant. The last modification to the contract was No. 8, dated June 16, 1967.

Dairyland Power Cooperative and the United States Atomic Energy Commission entered into a contract, AT(11-1)-851 on June 6, 1962, to buy steam from the nuclear power plant to operate a turbine-generator for production of electricity.

On November 5, 1962, Allis-Chalmers applied for a Construction Authorization.

The USAEC issued Construction Authorization, CAPR-5 on March 29, 1963.

On August 3, 1965, Allis-Chalmers applied for an Operating Authorization; amendments to the application continued through March 8, 1967.

The USAEC issued Provisional Operating Authorization No. DPRA-5 to Allis-Chalmers on July 3, 1967.

DPC applied for an Operating Authorization on October 4, 1967.

Provisional Operating Authorization No. DPRA-6 was issued to DPC on October 31, 1969, under Docket No. 115-5.

DPC applied to the USAEC to convert POA No. DPRA-6 to a 10 CFR Part 50 provisional operating license on May 22, 1972.

The USAEC issued Provisional Operating License No. DPR-45 under Docket 50-409 to DPC on August 28, 1973.

DPC applied to the USAEC to convert POL No. DPR-45 to a full-term facility operating license on October 9, 1974. The 40-year term would expire on March 28, 2003.

In 1977, the Systematic Evaluation Program (SEP) was initiated by the U.S. Nuclear Regulatory Commission (NRC) to review the designs of older operating nuclear power plants, including LACBWR, in order to reconfirm and document their safety. The purpose of the review was to provide (1) an assessment of the significance of differences between current technical positions on safety issues and those that existed when a particular plant was licensed, (2) a basis for deciding on how these differences should be resolved in an integrated plant review, and (3) a documented evaluation of plant safety. The conversion of the provisional operating license to a full-term operating license was tied to the completion of the SEP safety assessment. The Integrated Plant Safety Assessment for LACBWR was issued as NUREG-0827 in June 1983. Addendum 1 to NUREG-0827 was released in August 1986. DPC performed a consequence study to evaluate wind, tornado and seismic events. The study was accepted by the NRC in letters dated September 9, 1986 and April 6, 1987. DPC provided a schedule for completion of items necessary for safe shutdown during a seismic, wind or tornado event on December 11, 1986. Work on scheduled items has been terminated due to the plant shutdown.

2.3 OPERATING RECORD

La Crosse Boiling Water Reactor achieved initial criticality on July 11, 1967, and the low power testing program was completed by September 1967. In November 1967, the power testing program began. The power testing program culminated in a 28-day power run between August 14 and September 13, 1969.

Dairyland Power Cooperative has operated the facility as a base-load plant on its system since November 1, 1969, when the Commission accepted the facility from Allis-Chalmers.

The La Crosse Boiling Water Reactor was permanently shut down on April 30, 1987.

During this time the reactor was critical for a total of 103,287.5 hours. The 50 MW generator was on the line for 96,274.6 hours. Total gross electrical energy generated (MWH) was 4,046,923. The unit availability factor was 62.9%.

2.4 DECISION FOR SHUTDOWN

On April 24, 1987, the decision was made by the Dairyland Power Cooperative Board of Directors to permanently shut down LACBWR. The official announcement to the public of this decision was made on April 27, 1987.

The major reason for the decision was projected cost savings associated with the operation of the cooperative's coal-fired plants because of recently renegotiated coal and coal transportation contracts.

Other factors include the low growth rate in electrical demand forecast for the cooperative's service area and the current regional surplus of generating capacity.

Final reactor shutdown was completed at 0905 hours on April 30, 1987. The availability factor for LACBWR in 1987 had been 96.4%.

Final reactor defueling was completed on June 11, 1987. Eleven fuel cycles over the 20 years of operation have resulted in a total of 333 irradiated fuel assemblies being stored in the LACBWR Fuel Element Storage Well. Fifty-two new fuel assemblies remain in New Fuel Storage.

2.5 OPERATING EVENTS WHICH AFFECT DECOMMISSIONING

2.5.1 Failed Fuel

During refueling operations following the first few fuel cycles, several fuel elements were observed to have failed fuel rods. These fuel failures were severe enough to have allowed fission products to escape into the Fuel Element Storage Well and reactor coolant. These fission product particles then entered, or had the potential to enter and lodge in or plate out in, the following systems:

1) Forced Circulation
2) Purification
3) Decay Heat
4) Main Condenser
5) Fuel Element Storage Well
6) Overhead Storage Tank
7) Emergency Core Spray
8) Condensate system between main condenser and condensate demineralizer resin beds
9) Reactor Vessel
10) Seal Injection
11) Waste Water
12) Reactor Coolant Post-Accident Sampling System
13) Control Rod Drive System

Therefore, extra precautions will be taken in monitoring for and containing any fission product and transuranic radionuclide contaminants during the

eventual disassembly of the above listed systems. The majority of this material is located on horizontal surfaces in the Fuel Element Storage Well and the Reactor Vessel.

2.5.2 Fuel Element Storage Well Leakage

The stainless steel liner in the Fuel Element Storage Well (FESW) has had a history of leakage. From the date of initial service until 1980, the leakage increased from approximately 2 gallons per hour (gph) to just over 14 gph. In 1980, epoxy was injected behind the liner and leakage was reduced to approximately 2 gph. Since then, the leakage rate has slightly increased. Therefore, during SAFSTOR, the leakage rate will be periodically monitored. FESW water level will be continuously monitored in the control room and verified periodically by local inspection. The control room level instrument(s) will also generate an audible alarm when FESW level decreases to a selected level which is significantly above the minimum allowable level as specified in the technical specifications.

2.5.3 References

1) DPC Letter, LAC-4935, Madgett to Director of NRR, dated October 5, 1977.

2) DPC Letter, LAC-6274, Linder to Director of NRR, dated May 9, 1979.

3) DPC Letter, LAC-8553, Linder to Director of NRR, dated September 7, 1982.

3. FACILITY SITE CHARACTERISTICS

3.1 GEOGRAPHY AND DEMOGRAPHY CHARACTERISTICS

3.1.1 Site Location and Description of Site Layout

The La Crosse Boiling Water Reactor is located on the east bank of the Mississippi River approximately 19 miles south of the city of La Crosse, Wisconsin, and 1 mile south of the populated portion of the village of Genoa, Wisconsin. The site is, in the most part, owned by the Dairyland Power Cooperative and currently includes a mothballed 14 megawatt Genoa Unit 1 oil-fired generating facility, the 50 megawatt La Crosse Boiling Water Reactor and the 350 megawatt coal-fired generating facility, Genoa Unit 3. Plans exist to dismantle the oil-fired unit.

The site is on fill in the river-bottom area of the east bank of the Mississippi River and includes a portion of a wooded hillside to the east of the nuclear unit. The site also contains DPC's 161-KV and 69-KV transmission switching center and a distribution center for 34.5 KV.

The municipalities, including villages, towns and cities within a 25-mile radius of the facility, are shown in Figure 3.1. The population dispersion out to 5 miles is shown on Figure 3.2.

3.1.2 The Authority of the Exclusion Area and Licensee Authorities

The site exclusion area referenced in 10 CFR 100, Section 3(a) was initially established as approximately 1,109 feet in radius from the center of the Reactor Building. The area to which access will potentially be excluded during a postulated accident while in SAFSTOR is the area within the Effluent Release Boundary (ERB). The ERB is the Dairyland Power Cooperative property line within the former 1,109 ft. radius exclusion area. (See Figure 3.3.) DPC exercises direct control over its own employees and visitors on the site to exclude them if adverse radiological conditions require. Additionally, Dairyland Power Cooperative maintains a letter of agreement with the Vernon County Sheriff's Department for them to provide any necessary assistance.

3.2 TRANSPORTATION, INDUSTRIAL AND MILITARY FACILITIES WITHIN PROXIMITY TO THE PLANT

There are no military facilities located within a 5-mile radius of the nuclear plant site. The only industrial facility of any significant size is the Dairyland Power Genoa Unit 3 coal facility located approximately 200 feet from the nuclear plant and sharing the same site. There are no major manufacturing facilities of any type in this area; it is principally used for agriculture. Transportation routes include the Burlington Northern Railway line from Chicago, Illinois, to the West Coast which crosses through the original exclusion area. The Burlington Northern Railway line is a twin-track line of welded steel track and constitutes a major rail corridor for the railroad. Wisconsin State Trunk Highway 35 also crosses through the original exclusion area. The Mississippi River main channel which is used for barge transportation crosses through the original exclusion area. The

Milwaukee Railroad single track line from Minneapolis, Minnesota, to St. Louis, Missouri, is on the opposite side of the Mississippi River from the plant and was abandoned from 1980 to 1981. The line has since been restored to service but is not frequently used. State Trunk Highway 56 originates in the village of Genoa and runs East towards Viroqua, the county seat. The origin point for Highway 56 is approximately 1-1/2 miles north of the reactor plant.

On the Iowa and Minnesota side of the river, State Trunk Highway 26 runs within 4 miles of the original exclusion area. All the mentioned highway facilities are two-lane paved roadways with unlimited access.

The car count on the road (Highway 35) passing through the nuclear facility original exclusion area is 2,950 cars per 24 hours, as determined by the Vernon County Wisconsin Highway Department in 1984.

There does exist north of the plant, approximately .9 mile, a U.S. Army Corps of Engineers Lock and Dam on the Mississippi River. This lock is not classified as an industrial facility, although it employs approximately 11 individuals.

3.3 METEOROLOGY

3.3.1 Meteorological Measurement Program

The LACBWR meteorological measurement program consists of two levels of onsite equipment located within the Mississippi River valley. Surface (10m) meteorological parameters monitored are wind speed, wind direction, stability (sigma theta), and temperature. Stack (100m) meteorological parameters monitored are wind speed, wind direction, and stability (sigma theta and delta temperature). The 100m equipment is located on a boom at the top of the 100m LACBWR vent stack. Data is also available from the National Weather Service (NWS) station at the La Crosse Municipal Airport approximately 35 km (21.7 mi.) north of LACBWR.

3.3.2 General Climatology

The plant site area exhibits a typical continental type of climate. Temperature extremes in the La Crosse/LACBWR region are more marked because of the river-valley location. Average temperatures vary from -7.1°C (19.2°F) in the three months of winter to 21.9°C (71.4°F) in the summer months.[1] A maximum temperature of 42.2°C (108.0°) was recorded in July 1936, with a minimum low of -41.7°C (-43.0°F) recorded in January 1873, both in La Crosse. Monthly precipitation in the area averages between 5.1 cm (2.0 in.) and 10.7 cm (4.2 in.) from March through October and 2.5 cm (1 in.) and 5.1 cm (2 in.) for the rest of the year. Average annual precipitation is 79.2 cm (31.2 in.). Monthly snow and sleet averages between 12.7 cm (5 in.) and 35.6 cm (14 in.) from November through March, the largest amount normally occurring during March. The normal annual amount of snow and sleet is 110.5 cm (43.5 in.).

The prevailing winds are subjected to the channeling effects of the river valley. This channeling directs almost all of the regional scale cross-valley winds into the north-south orientation of the valley. Wind speeds are also lower as a result of vertical decoupling caused by the river valley. In summer, low wind speeds cause air stagnation in the valley during periods of hot humid weather, and in winter some deepening of inversion conditions can cause a stagnant layer of cold air on the valley floor.

3.3.3 Local Meteorology

Onsite meteorological data has been collected at two levels, on top of the vent stack 100m, and at the surface 10m, since 1976. In 1982 a new monitoring and data acquisition system was installed and the surface 10m tower moved to a location with better exposure. (See Figure 3.4).

Wind direction frequency distributions for the surface and stack levels are shown in Table 3-1. The distributions demonstrate the strong predominance of wind directions from the SSE and NNW sectors for the surface and S and N sectors for the stack. The Mississippi River valley has a north-south orientation at the plant site, and it would be expected that winds should be predominantly from the north and south because of the river valley's channeling effect. It is suspected that the layout of the buildings on site reduces the frequency of winds observed from the southwest to west.

TABLE 3-1

WIND DIRECTION FREQUENCY DISTRIBUTION
AT LACBWR SITE AND LA CROSSE NWS
(Percentage 1982-1984)

	Surface	Stack	East Bluff	LaCrosse NWS
N	10.4	15.9	5.6	5.5
NNE	2.1	5.1	4.7	1.9
NE	1.8	2.5	3.7	1.1
ENE	2.0	1.8	4.9	1.3
E	3.0	1.9	5.5	6.7
ESE	3.1	2.3	5.9	9.1
SE	6.1	5.2	6.2	10.6
SSE	20.9	10.2	7.5	10.9
S	14.1	22.5	10.8	12.1
SSW	3.4	6.6	9.5	3.4
SW	1.4	2.8	5.6	2.3
WSW	1.0	2.2	4.2	1.6
W	1.3	2.9	5.6	7.7
WNW	2.7	3.7	6.7	6.1
NW	9.3	7.2	6.8	10.9
NNW	17.7	7.3	6.8	8.6

The stack wind directions demonstrate a higher percentage of winds coming from the S to N directions than the surface distribution, due to the better exposure of the stack sensor from those directions. It is obvious that winds from the eastern and western sectors (at either level) are very infrequent because of the bluffs approximately 305m (1,000 feet) east of the site and the interference of buildings on the site to the west. The similarity between topographical features of the site and the La Crosse National Weather Service station are shown in the wind direction frequency distribution in Table 3.3-1. The major differences in the La Crosse distribution are due to the better exposure of the instrumentation at the airport, and LACBWR's proximity to the eastern bluff.

Monthly average temperature and wind speed data is presented for both the La Crosse and LACBWR sites in Figure 3.5. There is excellent correlation between the temperature records for both sites, with some small differences in the winter period. The difference for the months of December through February are most likely the result of a micrometeorological heat island effect at the LACBWR site. The warmer winter temperatures result from the influence that the large concrete buildings and roads have on local environment. Wind speeds for both the stack and the surface follow each other quite closely, with the stack wind speeds on the average approximately 65 percent higher than the surface wind speeds at the LACBWR site. La Crosse NWS wind speeds average 42 percent higher than site surface wind speeds. All locations also exhibit the typical case of higher wind speeds in the spring and fall than in the summer and winter months.

High wind speeds, in excess of 11.2 m/s (25 mph) are not prevalent at either the surface or stack locations, equaling 0.0 and 1.3 percent, respectively. Low wind speeds (Figure 3.6) are very prevalent at the surface site due to the sheltering effects of the nearby bluffs and buildings. Overall, there is very good agreement between LACBWR and La Crosse NWS wind speed, direction, and temperature data. The minor differences that do exist are due to differences in river valley influences and sheltering effects.

3.4 HYDROLOGY

3.4.1 Hydrologic Description

The reactor site is in the Mississippi River valley. In the vicinity of the site, the valley is deeply cut into highly dissected uplands. From La Crosse to Prairie du Chien, approximately 40 miles south, the valley varies between 2-1/2 and 4-1/2 miles in width. The valley walls rise sharply 500 to 600 feet from river level.

There is little or no agricultural use of the river valley floor which consists primarily of marshy land, islands between river channels and extensions of low lying flood plain cut by ponds, sloughs and meandering stream channels. Numerous short, steep-sided valleys that have been cut into the uplands by tributary streams intercept the main river valley. Both walls of the main channel are wooded. The flat upland areas and some of the tributary valleys are cultivated and grazed.

The main channel of the river varies greatly in width above and below the site. A series of dams are operated by the United States Army Corps of Engineers for navigational purposes. Above Dam No. 8 (about 3/4 mile north of the site) the river is nearly four miles wide. Below the site, the river is relatively narrow for a distance of 20 miles, then gradually widens as the river approaches Dam No. 9, 33 miles south of the site.

3.4.2 Drainage

The site is on a filled-in area south of the original Genoa steam plant. Therefore, drainage at the site must be provided. There is allowance for runoff from the high valley walls to the east. The site is favorably located with respect to this runoff, however, because of two short valleys east of the bluffs bordering the site. One valley drains to the north and one to the south, so that only precipitation that falls on the bluff adjacent to the site and on a small portion of the upland area contributes to runoff directly across the site. This runoff is presently channeled along the highway and railroad to prevent interference with traffic. No problems of flash floods have occurred at the site.

3.4.3 Downstream Water Use

For a distance of 40 miles downstream of the site, virtually all municipal water supplies for cities and towns along the river are obtained from ground water. On the basis of readily available published records, the nearest major city using the river water for direct human consumption is Davenport, Iowa, about 195 miles downstream. The nearest user of river water for industrial purposes, excluding the adjacent fossil plant, is the steam-power plant in Lansing, Iowa, about 15 miles downstream. River water is used at this plant for condenser cooling. There is no other known user of river water for industrial purposes between the reactor site and Prairie du Chien, 40 miles down-river.

3.4.4 Flooding and Probable Maximum Flood

The flood profile at the site of the La Crosse Boiling Water Reactor has a return frequency (as described by the U.S. Army Corps of Engineers) of 635.2 feet above main sea level (MSL) for a 50-year flood, 637.2 feet MSL for 100-year, and 640.0 for a 500-year. The site fill is at 639 feet. The Nuclear Regulatory Commission, during the Systematic Evaluation Program, determined that the maximum historic flood (1965) was 638.2 feet MSL. The standard project flood is 643.2 feet MSL and the probable maximum flood is 658 feet MSL. The period of record keeping for evaluation of flooding, in the region of the La Crosse plant, goes back to records kept by the United States Weather Bureau in La Crosse, Wisconsin, from approximately 1873 on. Site surface run-off flooding for a local probable maximum precipitation was determined to meet NRC criteria as the total run-off would be approximately 6.4 inches above grade and the equipment is protected to a level of approximately 1 foot or more above grade. This was in compliance with the applicable Regulatory Guide criteria based on a local run-off area of a 35-acre water shed to the east of the facility.

3.4.5 Potential Dam Failures

The U. S. Army Corps of Engineers maintains a lock and dam less than a mile upstream from the facility. The lock and dam was constructed in the 1930's. The NRC technical reviewers noted that Lock and Dam No. 8, upstream from the La Crosse Boiling Water Reactor, has its right bank earth-bermed to control water and direct flow to the dam spillway, which is located in the main river channel. Locks are located on the east bank adjacent to which is the U. S. Army Corps of Engineers' Field Office.

The failure of the main dam or adjacent earth-berms will have a variable effect on the water surface elevations at the La Crosse site. Barges depend on the river discharge for adequate channel depth. The nominal operating pool elevations of Lock and Dam No. 8 are 631 feet MSL (upstream) and 620 feet MSL (downstream). The difference in elevation between head and tail waters of the dam is 0.8 feet at the five-year discharge flow rate of 134,000 cfs. The elevation difference decreases with increasing discharge, so that at a 500-year discharge flow rate (321,000 cfs), the difference is reduced to 0.4 feet. Additional increases in discharge result in a smaller difference in elevation up to the elevation at which the dam is submerged.

Should the dam fail with discharges ranging from 100,000 cfs to 300,000 cfs, the increase in dam tail water elevations will be attenuated as water reaches the La Crosse site. Consequent increase of water elevation will certainly be less than 1 foot of elevation at the site. It was concluded that the effect of a catastrophic failure of Lock and Dam No. 8, during high flow conditions, would have negligible effect on water surface elevations measured at the La Crosse site.

3.4.6 Flooding Protection Requirements

Historical flooding protection at the La Crosse site was consistent with the initial design criteria of meeting the passive protection needs of a 100-year return frequency flood. These design criteria were based on the return frequencies established by the United States Geological Survey. During the Systematic Evaluation Program, the Nuclear Regulatory Commission reviewed, through a consultant, current criteria which establish a return frequency more in the approximate range of one in a million years. In this particular review, it was determined that in order to comply with criteria for a probable maximum flood certain evaluations would have to be completed. The existing reactor site did not comply with passive protection requirements to the level of one-in-a-million-year return frequency (probable maximum flood). The site was required to review the stability of certain structures at this flood level. It was determined that the reactor containment building and reactor stack would be able to withstand this flood. Procedures have been established which require certain actions to be taken at various water levels. A river elevation of 630 feet MSL activates the flood-alert stage in which management is alerted of the condition and monitoring is increased. A flood warning condition is declared at 635 feet MSL. At this point, flood control operations are coordinated with any offsite resources needed. Any

anticipated temporary dike construction is commenced depending on estimated final flood level. At 639 feet MSL, a flood emergency is declared. At 643 feet MSL a flood crisis level is declared. At this point, actions are taken to minimize the differential pressure on the containment vessel. The warning available to the facility of flood cresting is 4-5 days following crest at Minneapolis, Minnesota.

3.4.7 Ultimate Heat Sink and Low Flow Conditions

The ultimate heat sink of the La Crosse Boiling Water Reactor is the Mississippi River. Low flow to the site occurs in the fall and winter and the most frequently recorded lowest monthly average flow occurs in February. Minimum flows have also been recorded in August and September during periods of drought. Records of minimum and average flows maintained over the period of 1930 to 1955 at the United States Geological Survey Station at La Crosse were reviewed and are summarized as follows. These low flows should vary only slightly from those at the site.

Summary Flow Data for the Mississippi River at La Crosse Station 1930-1955:

Condition	Discharging Cu. Ft./Sec.
All Time Low Flow Rate December 30 and 31, 1933	3,200
Median of Annual Minimum Flow Rates (Averaged over 1 day)	8,100
Overall Average Flow Rate 1930-1955	27,970

As discussed in the Loss of Fuel Element Storage Well Cooling analysis (Section 9.4), the fuel pool can be without active cooling for 5 days before boiling commences. This time period will increase as the fuel continues to decay. Even if the Mississippi level falls below the Low Pressure Service Water Pumps suction for a longer period, 5 days provides sufficient time to set up a temporary cooling water supply.

3.4.8 Ground Water

As the site has valley sand overlaying a layer of Eau Claire sandstone of the Cambrian Age which is underlaid by a Mount Simon sandstone, wells have been driven in areas closest to the site but not in valleys characterized by sublayers of Mount Simon sandstone. Deep wells penetrating the Mount Simon layer flow to the surface indicating an artesian head above the level of the river valley floor. Use of water from these artesian aquifers has been limited because the chemical quality of this deep water is poorer than that from shallow aquifers. As a result, there has been no extensive withdrawal of water and no serious decrease in the artesian head. Therefore, an accidental release of contaminants cannot enter the artesian aquifer.

3.5 GEOLOGY, SEISMOLOGY, AND GEOTECHNICAL ENGINEERING

3.5.1 Basic Geologic and Seismic Information

The La Crosse Boiling Water Reactor (LACBWR) is located within the Wisconsin Driftless section of the Central Lowland physiographic province.[2] The Wisconsin Driftless section was not glaciated during the Pleistocene Epoch and is characterized by flat lying, naturally dissected sedimentary rocks of early Paleozoic age.[3] Moderate to strong relief has been produced on the unglaciated landscape which has been modified only slightly by a mantle of loess and glacial outwash in the larger valleys of the area. Maximum relief in the region is about 1000 feet.

Bedrock in the site region consists of Pre-Cambrian crystalline rocks exposed at the crest of the Wisconsin Dome by early Paleozoic (Cambrian and Ordovician, 572 million years before present [mybp] to 435 mybp) sedimentary strata. Basement rocks in the site vicinity are of granitic composition. The Paleozoic rocks are 1200-1300 feet thick in the site vicinity and consist of dolomites, sandstones and shales. About 600 feet of this sequence is exposed along the bluffs on both sides of the Mississippi River in the plant vicinity. Prior to the Pleistocene Epoch (more than 2 mybp) the river had carved a gorge as much as 150 to 210 feet deeper than can be seen today. It was buried by post-glacial sediment.

The site is located within the Central Stable Region tectonic province[4,5]. The Central Stable Region consists of a vast area of large circular uplifts and sedimentary basins, and broad synclines and arches. Major structural features include the Wisconsin Dome and Arch, Lake Superior syncline, Forest City Basin, Michigan Basin, and Illinois Basin. These structures were formed during the Late Pre-Cambrian and Early Paleozoic (more than 435 mybp).

Major uplift and downwarping also occurred during Late Paleozoic (330 mybp to 240 mybp). Some minor tilting occurred during and following the Pleistocene glaciation (2 mybp to 10,000 ybp). The site is located on the southwest flank of the Wisconsin Dome and the western flank of the Wisconsin Arch, a southward projection of the Wisconsin Dome. For this reason sedimentary strata in the site vicinity dips less than 20 feet per mile to the southwest.

Many faults have been mapped in the site region. None of these faults are considered to be capable according to 10 CFR Part 100, Appendix A. These faults are discussed in Section 3.5.2. A "capable fault" is a fault which has exhibited one or more of the following characteristics:

(1) Movement at or near the ground surface at least once within the past 35,000 years or movement of a recurring nature within the past 500,000 years.

(2) Macro-seismicity instrumentally determined with records of sufficient precision to demonstrate a direct relationship with the fault.

(3) A structural relationship to a capable fault according to characteristics (1) or (2) of this paragraph such that movement on one could be reasonably expected to be accompanied by movement on the other.

The site lies on the east bank of the Mississippi River. Local drainage has dissected the upland areas into a pinnate pattern. The site is north of one of the drainages (Italian Hollow) which opens into the Mississippi Valley perpendicularly from the east. The Mississippi Valley is broad (2.6 miles at the site) and is bounded on both sides by vertical bluffs several hundred feet high composed of relatively flat lying early Paleozoic strata (570 mybp to 410 mybp).

The plant is founded on about 15 to 20 feet of hydraulic fill on the river flood plain. Beneath the fill are from 115 to 135 feet of fine sand that was deposited as alluvium and glacial outwash. The bedrock below these glacial fluvial deposits is sandstone of the Dresbachian Group of Cambrian age (570 mybp to 500 mybp).

3.5.2 Proximity of Capable Tectonic Structures in the Plant Site Vicinity

The NRC has been sponsoring research programs since the early 1970's in an attempt to determine if there are bases for delineating seismotectonic provinces and earthquake source structures in the eastern and central United States. Much new geologic and seismic information has been developed as a direct result of these studies, but positive evidence substantiating province boundaries and specific earthquake generating tectonic structures has not been found. The most successful results have been attained in the Mississippi Embayment region where a reactivated Precambrian rift zone (Reelfoot Rift) is indicated to be responsible for the relatively high seismicity there. Some success has also been achieved with respect to the Central Stable Region (site province) in developing seismic and geologic evidence that suggests the presence of seismic source zones[6]. The study, based on an independent examination of large geologic structures that appear to be spatially related to seismicity, proposes ten seismic source zones. These zones are related to either basement rifts or to major uplifts and basins. The nearest of these major zones to the LACBWR is the Cincinnati Arch, the northwest segment of which (Kankakee Arch) trends to within 200 miles southeast of the site. There is no geologic evidence, however, that any of these major tectonic structures (except for the Reelfoot Rift, also called the New Madrid Fault zone) are capable or that they are associated with capable structures.

Numerous faults have been mapped in the site region. Many investigations have been carried out concerning these faults by the state geological surveys, oil companies, mining companies, and consultants to utilities and agencies constructing nuclear facilities. Geologic history interpreted from all of these studies indicates that the last major tectonic activity took place during the period between post-Pennsylvanian (290 mybp) and pre-Cretaceous (138 mybp). The general absence of middle to late Paleozoic (410 mybp to 330 mybp), Mesozoic (240 mybp to 36 mybp) and Cenozoic (63 mybp to 2 mybp) strata make it extremely difficult to pinpoint the age of last

movement along these faults. However, Pleistocene (younger than 2 mybp) deposits are relatively common in the region except in the Wisconsin Driftless Area. Investigations in the region have not found evidence that Pleistocene deposits are offset nor that accumulations of Pleistocene deposits are thicker on the down thrown side of faults than on the up thrown side, even though such characteristics have been extensively searched for. Additionally, drainage patterns in the region that trend across mapped faults show no offset along the fault traces. Therefore, the faults are at least pre-Pleistocene and not capable according to Appendix A of 10 CFR Part 100.

The following is a brief discussion of each of the more significant faults in the site region within a radius of 200 miles.

The closest mapped fault of any size to the LACBWR site is the Mifflin fault, which is located on the northeast flank of an anticline, the Mineral Point anticline. It is located in Iowa and LaFayette Counties, Wisconsin, about 45 miles southeast of the site. The fault strikes N 40° W for about 10 miles and is offset with the northeast side down at least 65 feet. About 1000 feet of strike-slip displacement has also been indicated. Last movement on the fault is believed to have occurred in Late Paleozoic (330 mybp to 240 mybp)[7].

The Sandwich fault is a major regional fault in northern Illinois about 150 miles southwest of the site. The Sandwich fault is an 85-mile long vertical fault that strikes northwest-southeast, and is down to the northeast a maximum of 900 feet[8]. A subsidiary fault is present near the north end of the Sandwich fault with 150 feet displacement down to the south, forming a graben between the two faults. The nearest age of last movement that can be determined is post-Silurian and pre-Pleistocene as no intervening age rocks are present in association with the fault. During subsurface investigations for an expansion of the General Electric Company's Fuel Recovery Operation near the Dresden nuclear site, a complex fault zone was found. This fault zone was considered to be related to the same tectonic events that caused deformation on the Sandwich fault zone. These investigations showed that the fault zone at the GE facility did not offset the Pennsylvanian Spoon formation, thus demonstrating that last movement occurred more than 280 million years ago[9].

The Plum River fault zone (formerly the Savanna fault) is located 120 miles south-southeast of the site in northwestern Illinois and eastern Iowa. It strikes east-west and is due west of the northern end of the Sandwich fault. The fault zone consists of a series of echelon faults with south sides up from 100 to 400 feet. This fault zone is associated with the Savanna anticline, a major fold in the region. Last movement on this fault zone took place between post-middle Silurian (425 mybp) and pre-middle Illinoian (700 thousand years bp)[10].

The Madison fault and the Janesville fault are about 50 miles southeast of the site. They trend parallel in a general east-west strike with the Madison fault being the northernmost. The north side of both faults is down relative to the south side. The Janesville is the most well known of the two and has been mapped in the subsurface for a distance of 75 miles[11]. It is composed

of two branches, the predominant east-west one, and another that strikes northeast-southwest. Both faults are interpreted to have last moved sometime between post-Silurian and pre-Cretaceous (410 mybp to 138 mybp).

The Appleton and Green Bay faults are located 180 and 130 miles northeast of the site. Both faults have been postulated based on abrupt changes in elevation (south sides down) on the Precambrian basement[11]. The faults are dated as post-Silurian to pre-Cretaceous (410 mybp to 138 mybp).

Several faults are mapped in the Precambrian basement in Minnesota and northern Wisconsin based on geophysical evidence. These faults, in general, are oriented in a northeast to north-northeast direction. Two of these faults are the Douglas and Lake Owen faults which bound the north and south flanks of the Lake Superior syncline respectively. A southwest extension of the Lake Owen fault is the Hastings fault which approaches to about 110 miles from the site. Due to the lack of post-Precambrian rocks over these faults, it is impossible to determine an upper limit of last movement. They probably experienced activity during late Precambrian (600 mybp) and throughout the Paleozoic (570 mybp to 240 mybp). There is no evidence that they are capable.

Faulting has been identified in the Chicago area. A fault zone is inferred in the basement rocks from gravity and seismic data[12], north of, and parallel to, the Sandwich fault zone. The south side is down relative to the north side. It is not known whether or not the fault extends into the overlying Paleozoic rocks. Twenty-five minor faults are mapped in the Chicago area based on seismic data[13]. These faults are dated as post-Silurian (410 mybp) and pre-Pleistocene (2 mybp). There is no surface evidence for these faults, but from the seismic data, displacements of up to 55 feet are recognized. The fault zone which trends west-northwest consists of blocks that have been downdropped both to the north and to the south within the zone.

The LaSalle anticlinal belt trends along the eastern flank of the Illinois Basin to within 200 miles of the site. Faults have been postulated[14] on the west flank of the LaSalle anticlinal belt, the Oglesby and Tuscola faults. There is no direct evidence for faulting but they are based on the presence of changes in dip of the rock strata and as much as 1200 feet of stratigraphic difference on the west side of the anticline. Movement on these faults, if they exist, is interpreted to be pre-Cretaceous (more than 138 mybp).

The nearest region containing possible known capable faults is the Mississippi Embayment or New Madrid fault systems which is about 370 miles from the site. Extensive investigations in this area by the USGS, local universities, and state geological surveys, funded in part by the NRC, have indicated that recent faulting and earthquake activity are related to the reactivation of a north-south striking Precambrian and Paleozoic rift zone. Although seismicity of the New Madrid fault system must be considered in the seismic analysis of the LACBWR plant, it is not significant, by virtue of its distance from the site, in a consideration of potential surface faulting at the site.

Little is known about faulting in the rock beneath the site, but no faults have been mapped in the 400 to 600 feet high bluffs immediately east of the site. It is possible that minor faults mapped in the lead-zinc mining district southeast of the site are representative of faulting in the immediate site vicinity, although the lead-zinc district is in an area of relatively strong tectonic deformation. It is possible that minor faults are present in rock beneath the site, but based on low seismicity, a lack of any indication of fault displacements in outcrops in the area, and the lack of evidence for recent fault displacement in the region, it is concluded that faults beneath the site, if they exist, are not capable.

Conclusion of U. S. Nuclear Regulatory Commission

"There are no geologic conditions in the site vicinity that represent hazards to the facility. Numerous faults are mapped in the site region, but investigations of all of these faults during the course of validating several nuclear power plant sites in the region, in addition to studies for the LACBWR, have not found any evidence of capable faulting. Additionally, the area is one of relatively low seismicity. Therefore, capable faulting does not need to be considered in the analysis of this site."

3.5.3 Surface Faulting at the Site

The LACBWR is located one mile south of Genoa, Vernon County, Wisconsin, on the east bank of the Mississippi River. The LACBWR facilities are situated on about 15 feet of hydraulic fill which overlies approximately 100-130 feet of glacial outwash and fluvial deposits. Due to the absence of bedrock exposures at the plant site, the geologic investigation was restricted to an examination of the rock-bluffs in the site vicinity. The following observations were compiled from several vantage points:

- Bluffs on both sides of the Mississippi River are heavily vegetated;
- Scattered outcrops are visible;
- Bluff tops appear to be sub-horizontal at a relatively constant elevation;
- Valleys on one side of the Mississippi River do not appear to be linearly continuous with valleys on the opposite side;
- Several widely spaced joints are visible;
- A lithologic contact (bluff to light gray sandy dolomite to dolomitic sandstone overlying yellow-brown sandstone with gray siltstone interbeds) could be discontinuously observed in the bluff face immediately east of the LACBWR. This contact appeared to be sub-horizontal and was not observed to be off-set by folding or faulting; and

- No closely-spaced joints, shear zones, or faults were observed in the bluffs east of the LACBWR or in either valley immediately north and south of the plant.

Conclusions

Based upon the preceding observations, there is no apparent evidence to indicate that faulting has affected the Late Cambrian-Ordovician age rocks exposed in the bluffs east of the LACBWR. It is therefore apparent, based upon the available evidence, that there are no faults in the vicinity of the LACBWR that have the potential to represent a seismic hazard to site safety.

3.5.4 Stability of Slopes and Properties of Subsurface Materials

3.5.4.1 Properties of Subsurface Materials. The initial soil investigations at the La Crosse site were conducted in 1962. Between 1962 and 1980 soil test borings were made at 36 locations in the site vicinity. Of this number, five were associated with subsurface investigations in the power station area, four were associated with the switchyard area, and one was drilled to locate an offsite borrow area for construction fill materials. The remaining 23 were associated with subsurface investigations in the main plant facility area. DPC has boring logs depicting the soil conditions encountered in these investigations. Field investigation effort included standard penetration tests (SPT) and split-barrel sampling in accordance with ASTM D-1586-67 procedures. Relatively undisturbed samples were also obtained at several locations in thin-walled tubes using an Osterberg piston sampler. Laboratory testing of soil samples was accomplished to determine index properties and to establish soil strength parameters. Testing included specific gravity determinations in accordance with ASTM D-854-58, particle size analysis testing in accordance with ASTM D-422-63, relative density determinations in accordance with ASTM D-2049-69 and cyclic triaxial testing in accordance with the procedures of NUREG-0031.

3.5.4.2 Plant Facilities. The reactor containment structure, turbine building, diesel generator building, stack, waste disposal building and the gas vault are supported on cast-in-place concrete piles driven to develop a 50-ton capacity and filled with concrete specified to have a minimum 28-day compressive strength of 3500 psi. Using the data presented, the NRC staff independently estimated the bearing capacity of the in-place piles. Results indicate that the piles can be expected to safely carry a loading of greater than 50 tons per pile without significant settlement under static loading conditions. The cribhouse and associated water intake and discharge piping are not designated as seismic Category I structures.

3.5.4.3 Slope Stability. Review of available onsite and offsite topographic data indicates there are no onsite slopes whose failure could cause radiological consequences adversely affecting the public health and safety. One offsite slope, the east bank of the Mississippi River adjacent to the plant cribhouse site, was identified from topographic data and evaluated for safety in conjunction with this topic. A generalized typical section for this slope is presented in Figure 3.7.

A stability analysis of this slope was performed using conservative solid strength parameters developed from the results of the site subsurface investigation. In the analysis an angle of internal friction of 27° was assigned each soil layer and the slope stabilizing influence of the slope protection riprap material was conservatively ignored. Results of the analysis indicates the factor of safety against failure under static loading conditions is greater than 1.5. The U. S. Nuclear Regulatory Commission has concluded that an adequate margin of safety exists under static loading conditions.

Due to the fact that the cribhouse and the associated water intake and discharge piping are the only plant structures in the vicinity which will potentially be affected by a postulated failure of the east river bank slope, and these structures are not designated as seismic Category I structures, a dynamic (pseudostatic) slope stability analysis is not appropriate and was not performed.

3.5.4.4 Conclusions of U. S. Nuclear Regulatory Commission. Based on the review of available site data and on information obtained during an NRC staff visit to the site, it was concluded that the stability of slopes associated with the La Crosse Boiling Water Reactor site does not pose a safety concern for this plant.

3.5.5 Stability of Subsurface Materials and Foundations

3.5.5.1 Liquefaction and Seismic Settlement. The safe shutdown earthquake (SSE) peak ground acceleration postulated for the La Crosse site is 0.11g with an equivalent duration (NEQ) of 5 cycles. Results of standard penetration tests (SPT) undertaken by DPC in 1980 show a range in N-values in clean sand below the water table beneath the turbine building of from 12-34 blows/ft. SPT N-values taken beneath the stack ranged from 23 to over 50 blows/ft. Based on the NRC staff's review of the site foundation conditions, the borings under the turbine and stack foundations are considered representative for other adjacent structures that are pile supported including the reactor containment building.

Results of an NRC staff safety evaluation concerning liquefaction potential at the La Crosse site were reported in August 1980. Based upon an evaluation of information provided by DPC, the staff concluded in that report that the materials under the existing turbine building, stack, and the reactor containment structure are adequately safe against liquefaction effects for an earthquake up to a magnitude 5.5 with a peak ground acceleration of 0.12g.

Based upon the information presented by DPC that all plant seismic Category I structures are supported upon piles and the results of the NRC staff's previous studies, the staff concluded that induced settlements of seismic Category I structures would not be significant under the postulated dynamic conditions.

3.5.5.2 Turbine Building Floor Support Grouting. In July 1980, borings drilled under the turbine building (Borings DM-12 and DM-13) encountered voids at several locations beneath the "on grade" concrete floor slab. In order to identify the lateral and vertical extent of the voids and to investigate potential for voids under other safety-related structures, DPC accomplished an exploratory drilling program at the site. Results of the program indicated voids ranging in depths up to 10 in. existed only within the turbine building area. Although voids of this relatively small size would not significantly affect lateral support to the 310 fifty to seventy ft. long piles supporting the turbine building or the integrity of the overall turbine building pile foundation under dynamic loading conditions, DPC accomplished an injection grouting program to fill the voids and restore continuous "on-grade" support to the turbine building concrete floor. About 460 cu. ft. of grout was injected under a floor area of approximately 10,000 sq. ft. The grouting program was completed on October 28, 1980.

3.5.5.3 Conclusions of U. S. Nuclear Regulatory Commission. "Based on the review of licensee's submittal and of other available referenced data, the NRC staff concurs with DPC's conclusion that the pile supported structures, systems and components are not expected to experience excessive settlements under static or dynamic conditions."

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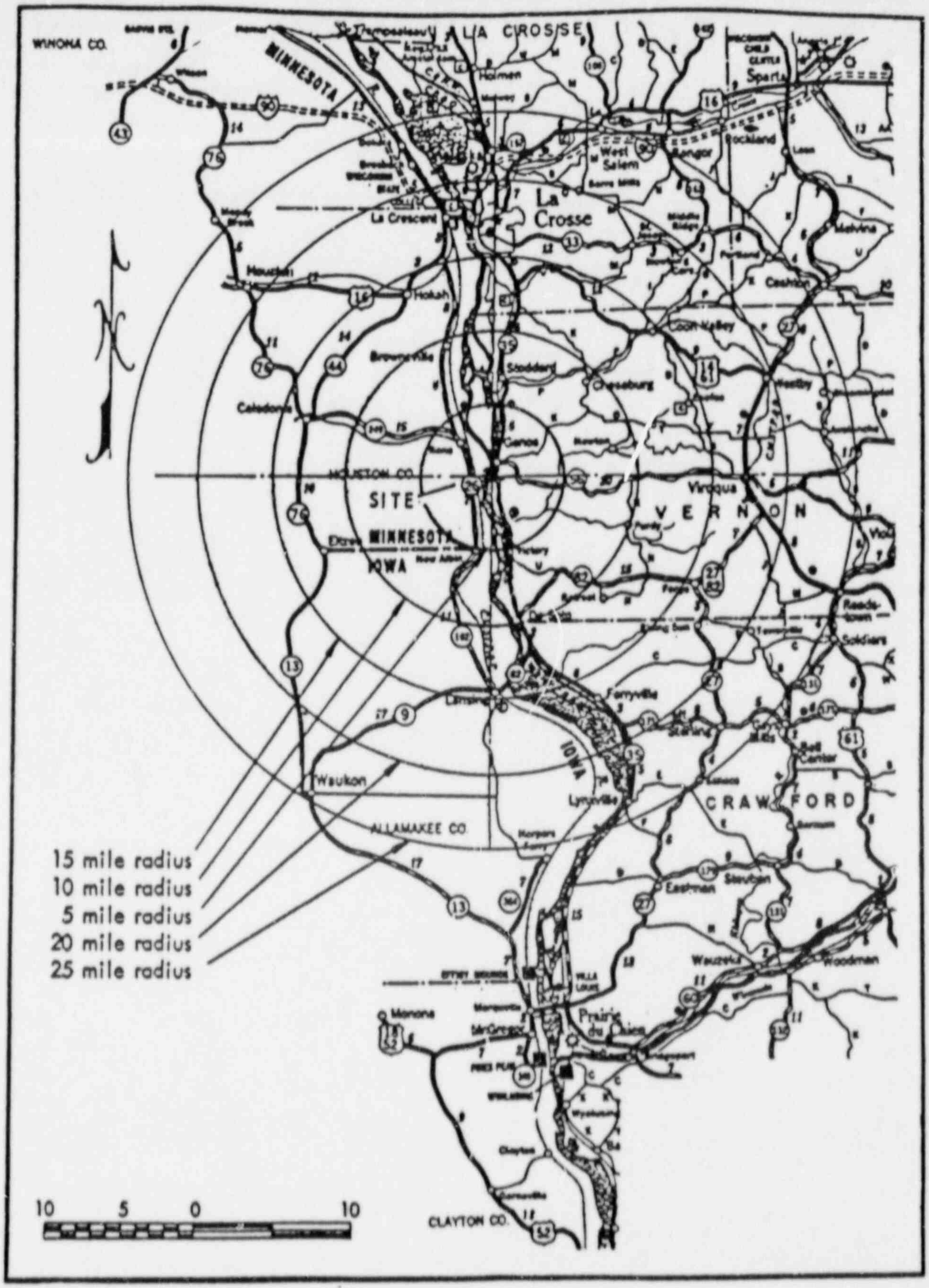
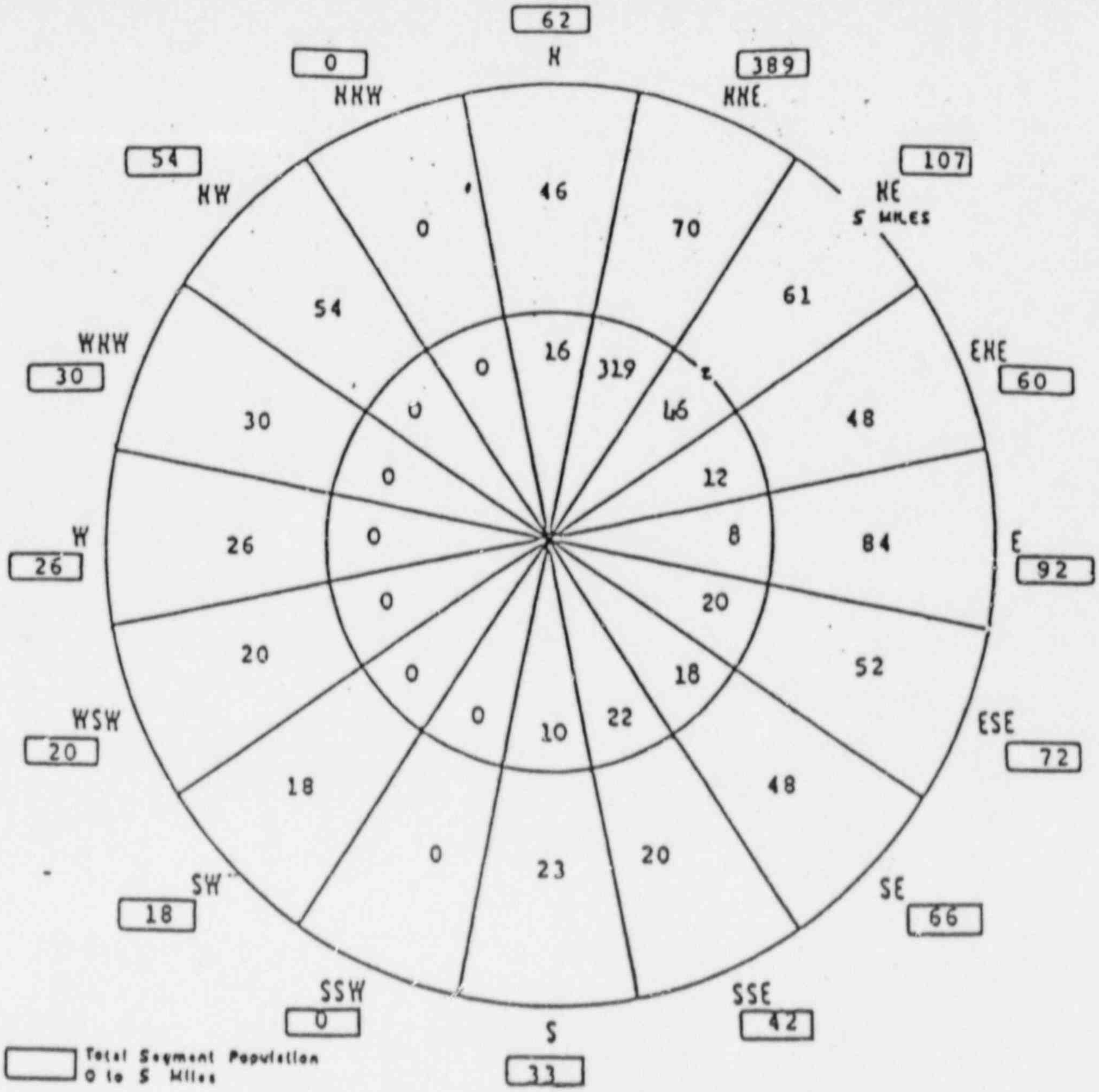


General Site Location Map

FIGURE 3.1



POPULATION TOTALS			
RING, MILES	RING POPULATION	TOTAL MILES	CUMULATIVE POPULATION
0-2	471	0-2	471
2-5	600	0-5	1071

Estimated

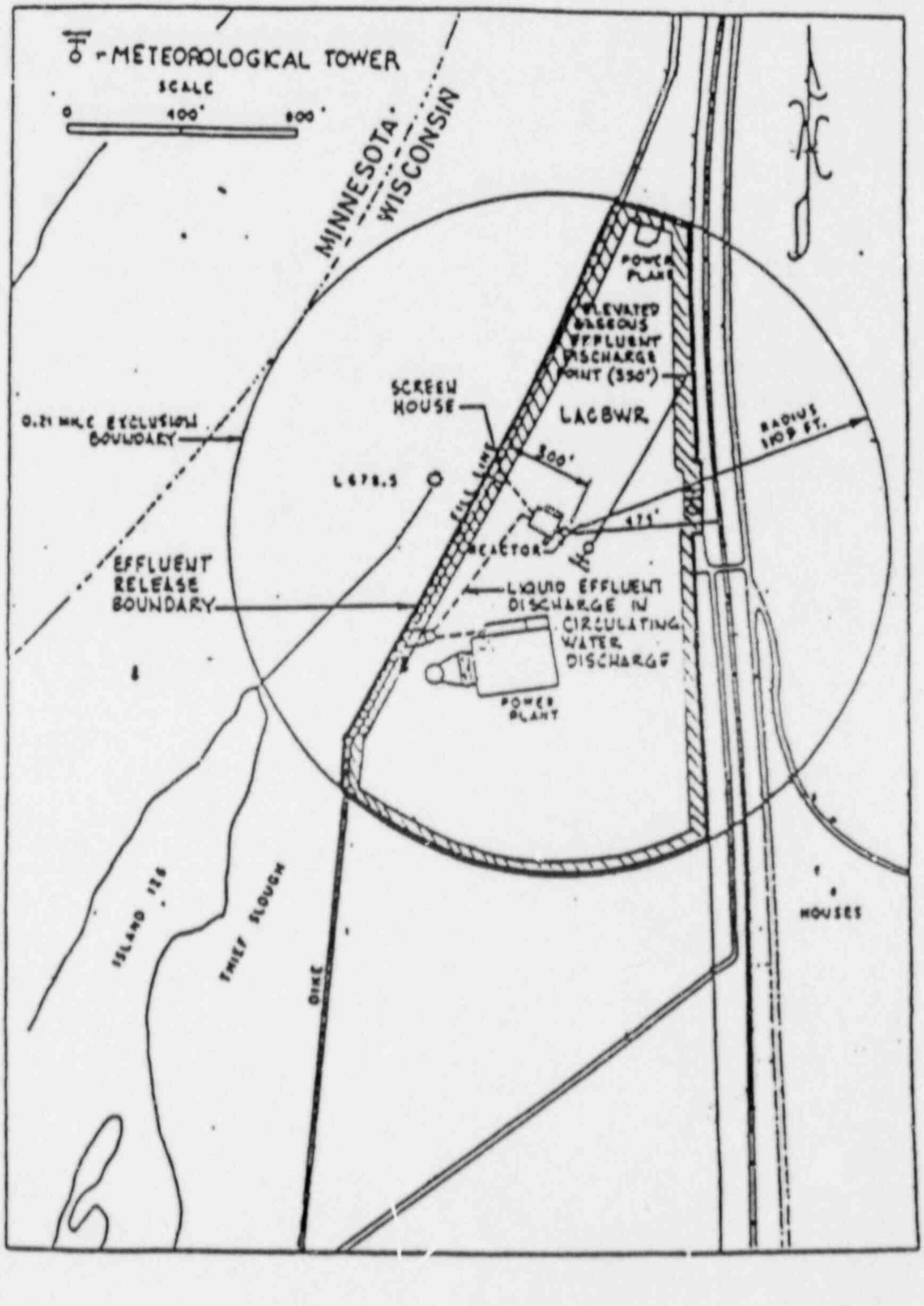
Permanent Population (6/82)

MN = 148 (13.82%) WI = 923 (86.18%)

IA = 0

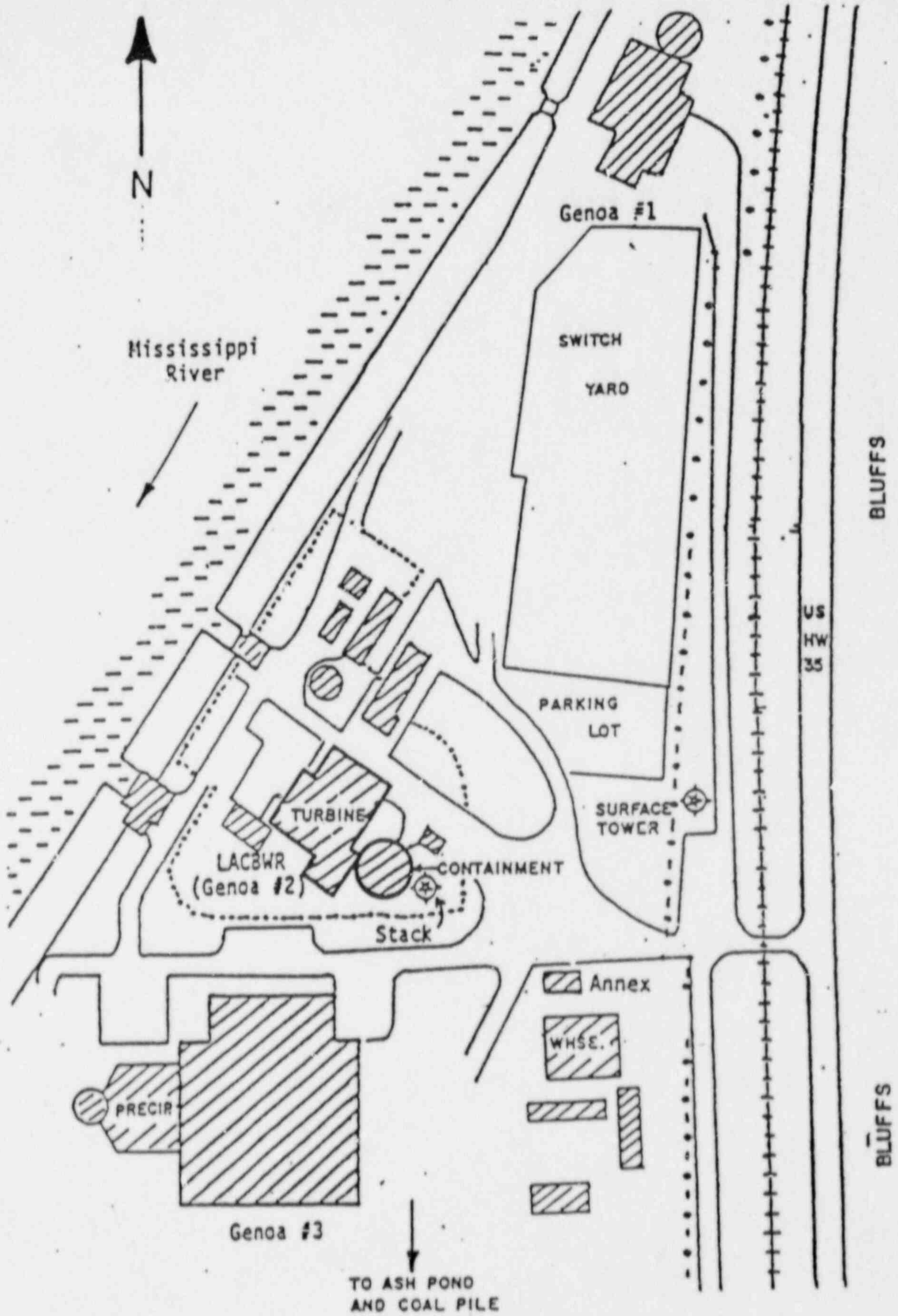
Population Dispersion

FIGURE 3.2



Effluent Release Boundary

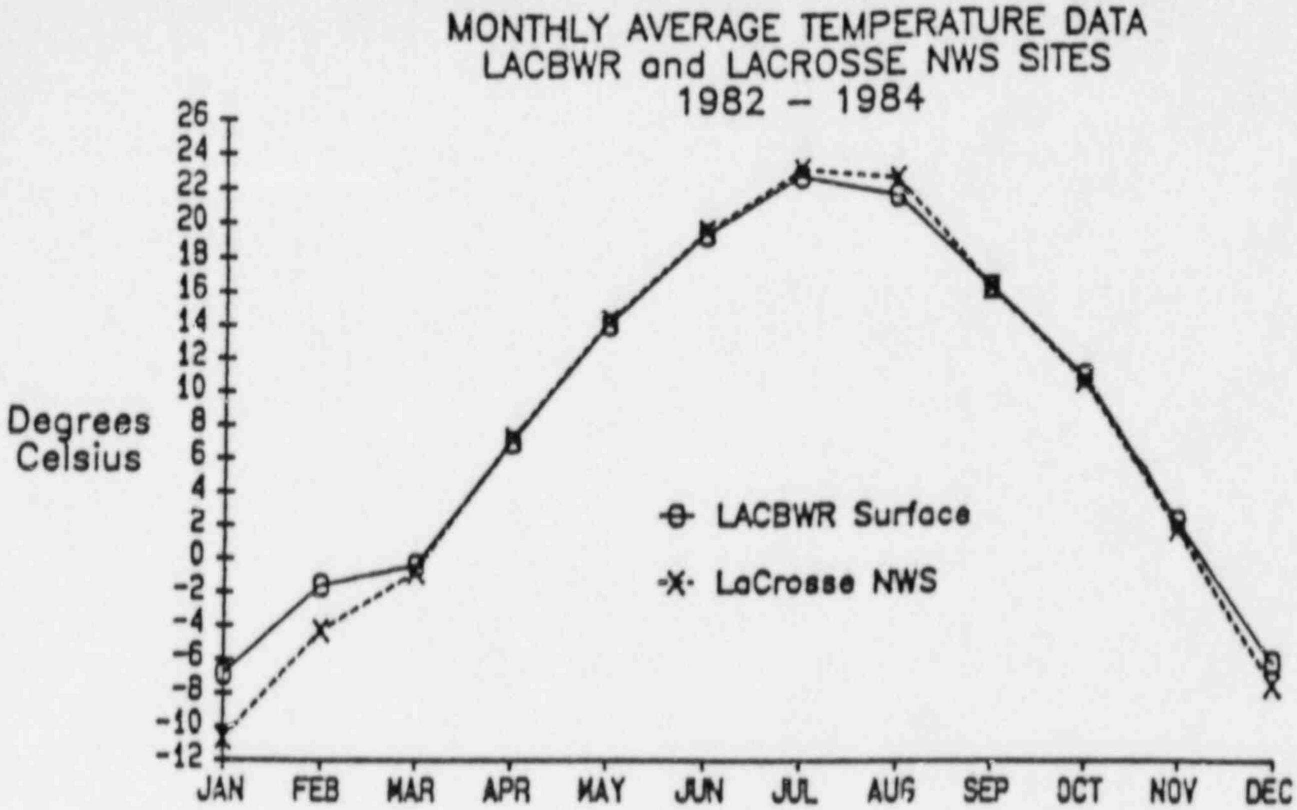
FIGURE 3.3

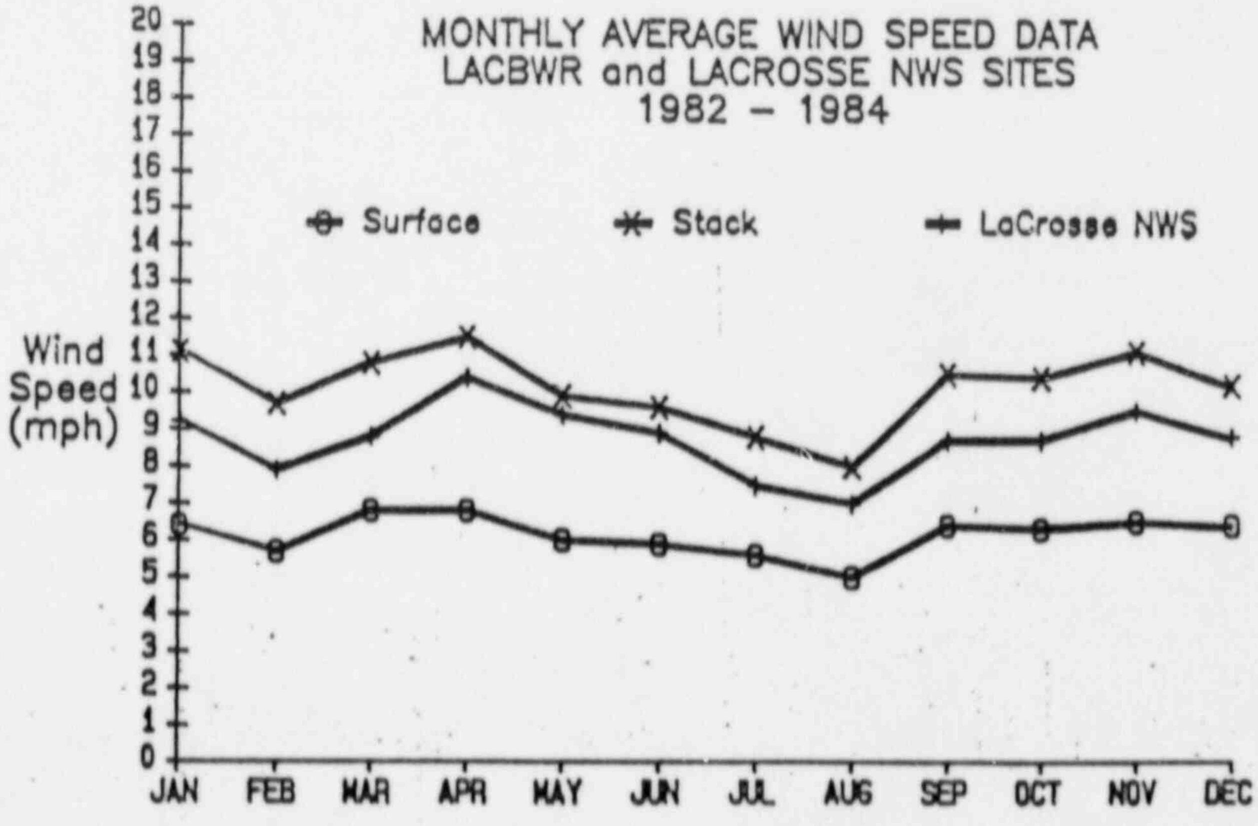


LACBWR Site Map

Scale: 1" = 185'

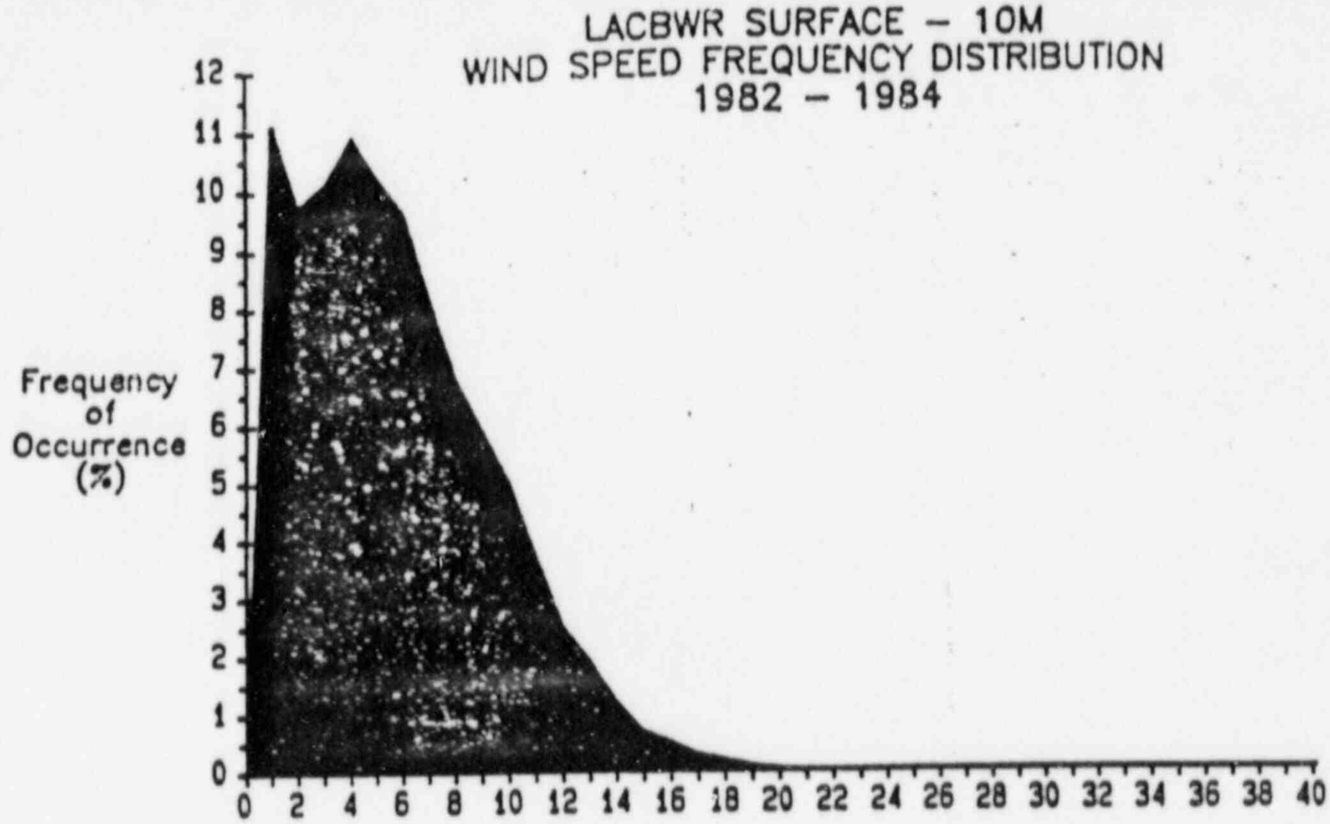
FIGURE 3.4

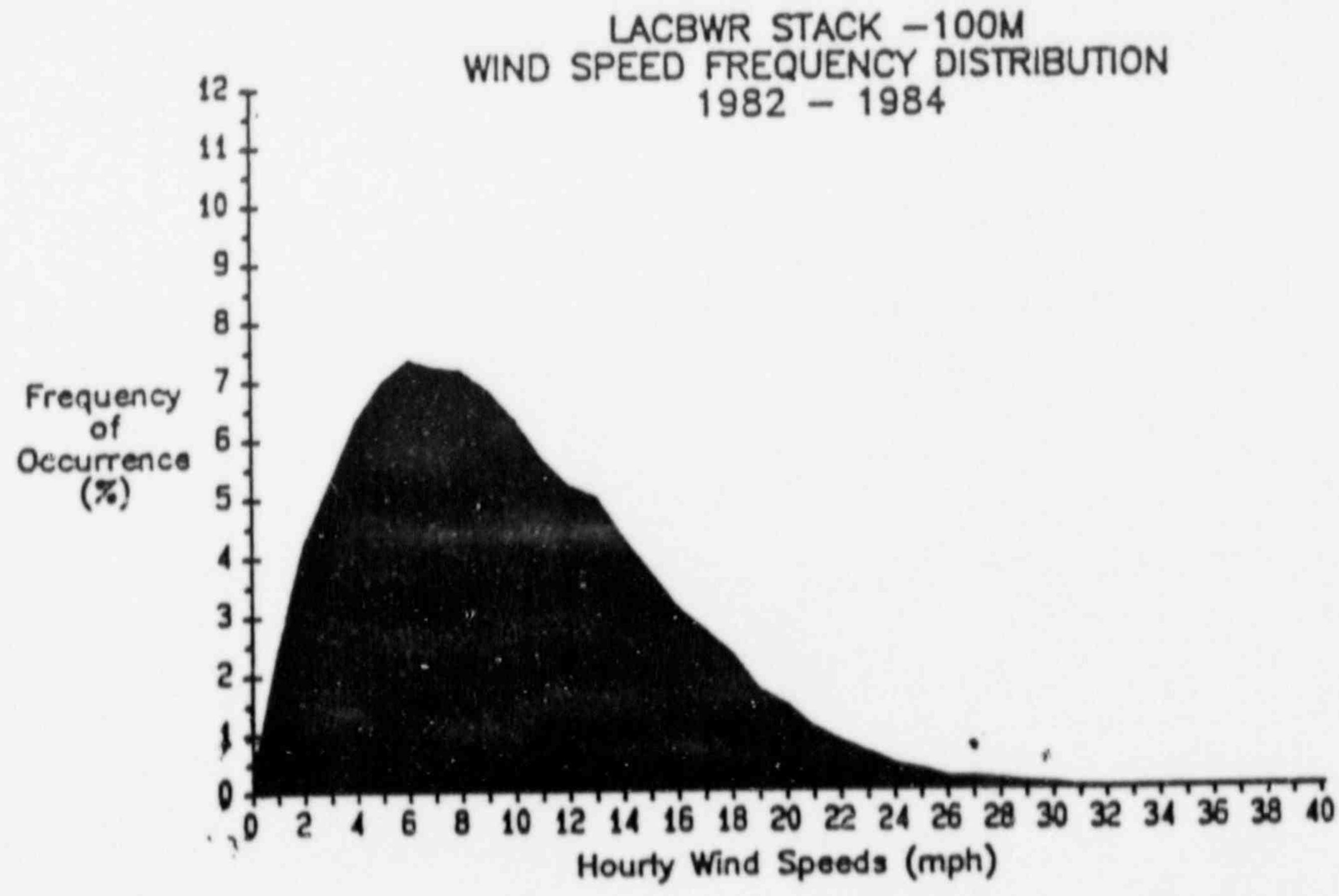




Monthly Average Meteorological Data

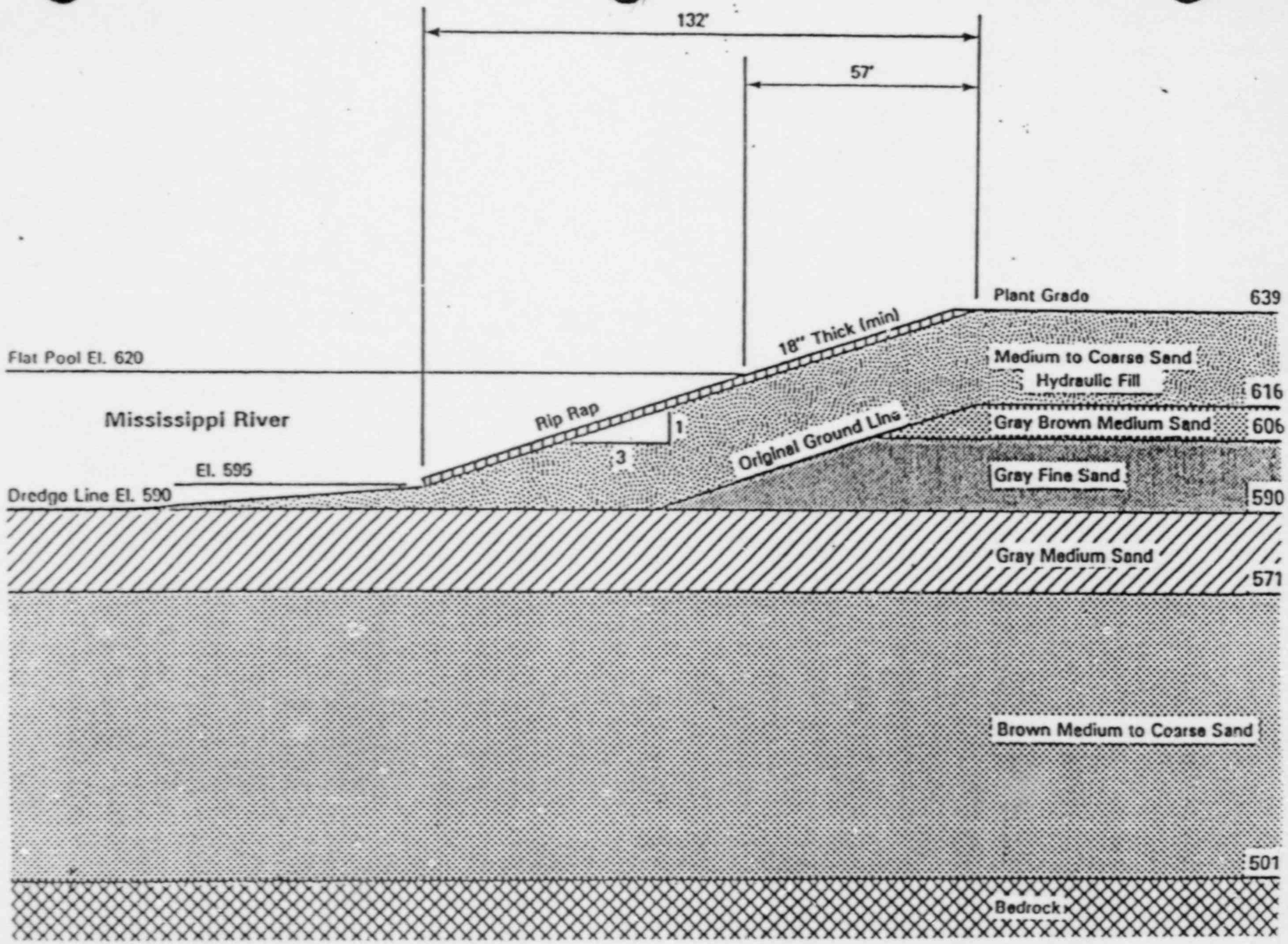
FIGURE 3.5





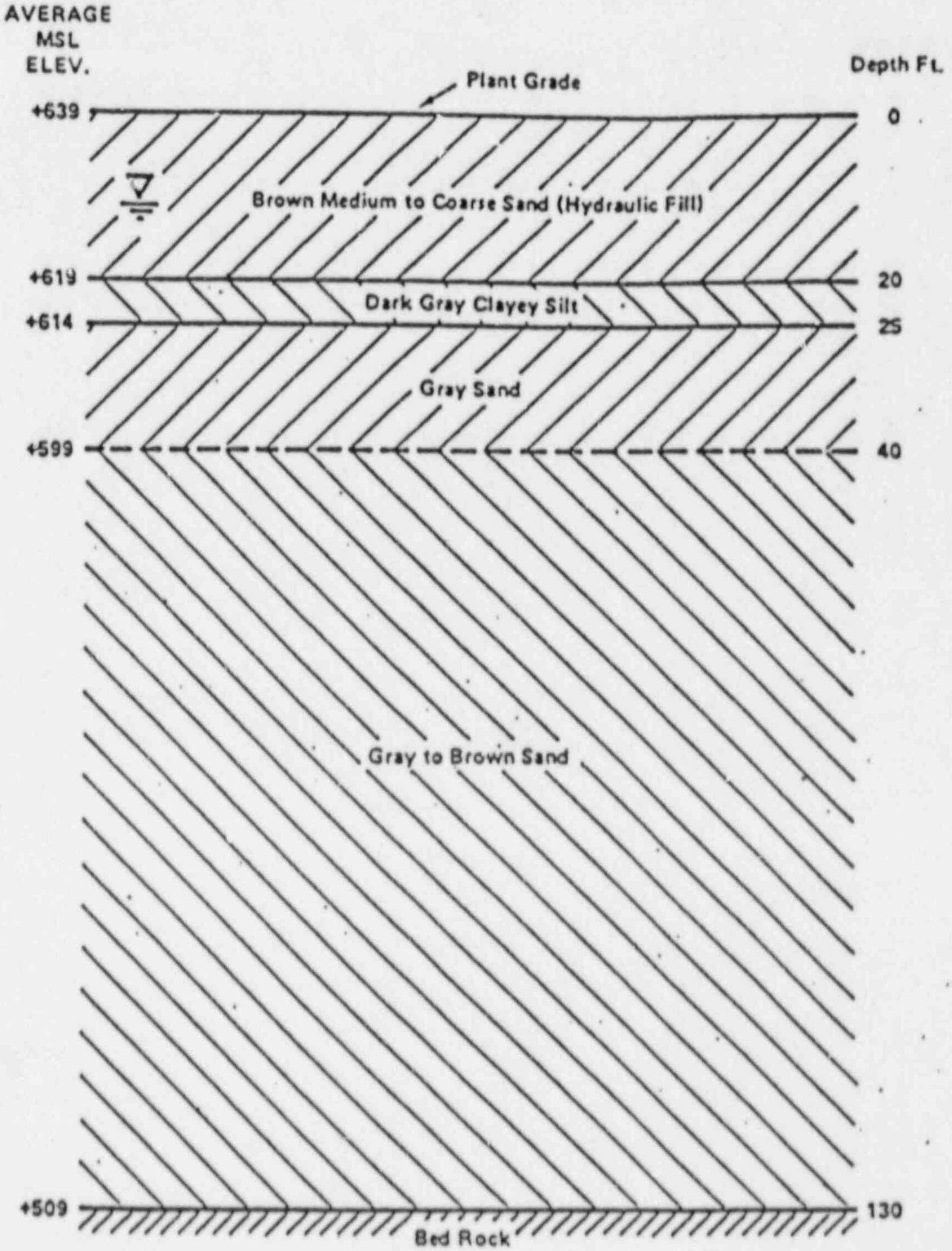
<u>Wind Speed Frequency Distribution</u>

FIGURE 3.6



East Bank River Slope

FIGURE 3.7



Generalized Soil Profile

FIGURE 3.8

4. FACILITY DESCRIPTION

4.1 GENERAL PLANT DESCRIPTION

The LACBWR was a nuclear power plant of nominal 50 Mw electrical output, which utilizes a forced-circulation, direct-cycle boiling-water reactor as its heat source.

The reactor and its auxiliary systems are within a steel containment building. The turbine-generator and associated equipment, the control room for both turbine and reactor controls, and plant shops and offices are in a conventional building adjacent to the containment building.

Miscellaneous structures which are associated with the power plant, and are located adjacent to the Turbine Building, include the electrical switchyard, Cribhouse, Waste Treatment Building, LSA Storage Building, oil pump house, stack, warehouses, administration building, annex building, guard house, outdoor fuel oil and acid storage tanks, underground septic tanks, gas storage tank vaults, underground oil tanks and the condenser circulating water discharge seal well at Genoa Unit 3.

Miscellaneous onsite improvements include roads, walks, parking areas, yard lighting, fire hydrants required for plant protection, access to and use of rail siding facilities, fencing, landscaping, and communication services.

4.2 BUILDING AND STRUCTURES

4.2.1 Containment Building

The containment building (Figs. 4.1 and 4.2) is a right circular cylinder with a hemispherical dome and semi-ellipsoidal bottom. It has an overall internal height of 144 ft. and an inside diameter of 60 ft., and it extends 26 ft. 6 in. below grade level. The shell thickness is 1.16 in., except for the upper hemispherical dome which is 0.60 in. thick.

The building contains most of the equipment associated with the nuclear steam supply system, including the reactor vessel and biological shielding, the fuel element storage well, the forced circulation pumps, the shutdown condenser, and process equipment for the reactor water purification system, decay heat cooling system, shield cooling system, seal injection system, emergency core spray system, boron injection system, and storage well cooling system.

The containment building was designed to withstand the instantaneous release of all the energy of the primary system to the containment atmosphere at an initial ambient temperature of 80°F, neglecting the heat losses from the building and heat absorption by internal structures. Its design pressure is 52 psig, compared to a calculated maximum pressure buildup of 48.5 psig following the maximum credible accident while in operation. The containment building shell is designed and constructed according to the ASME Boiler and Pressure Vessel Code, Sections II, VIII, and IX, and Nuclear Code Cases 1270N, 1271N, and 1272N.

The interior of the shell is lined with a 9-inch-thick layer of concrete, to an elevation of 727 ft. 10 in., to limit direct radiation doses in the event of a fission-product release within the containment building.

The containment building is supported on a foundation consisting of concrete-steel piles and a pile capping of concrete approximately 3 ft. thick. This support runs from the bottom of the semi-ellipsoidal head at about el. 612 ft. 4 in. to an elevation of 621 ft. 6 in. The 232 piles that support the containment structure are driven deep enough to support over 50 tons per pile.

The containment bottom head above el. 621 ft. 6 in. and the shell cylinder from the bottom head to approximately 9 in. above grade elevation (639 ft. 9 in.) are enveloped by reinforced concrete laid over a 1/2 in. thickness of premolded expansion joint filler. The reinforced concrete consists of a lower ring, mating with the pile capping concrete. The ring is approximately 4-1/2 ft. thick at its bottom and 2-1/2 ft. thick at a point 1-1/2 ft. below its top (due to inner surface concavity). The ring then tapers externally to a thickness of 9 in. at the top (el. 627 ft. 6 in.) and the 9 in. thickness of concrete extends up the wall of the shell cylinder to 639 ft. 9 in. The filler and concrete are not used, however, where cavities containing piping and process equipment are immediately adjacent to the shell.

Except for areas of the shell adjacent to other enclosures, the exterior surface of the shell above el. 639 ft. 9 in. is covered with 1-1/2-inch-thick siliceous fiber insulation, faced with aluminum. The insulation of the dome is Johns-Manville Spintex of 9 lb/ft^3 density, faced with embossed aluminum sheet approximately 0.032 in. thick. The insulation of the vertical walls is Johns-Manville Spintex of 6 lb/ft^3 density, faced with corrugated embossed aluminum sheet approximately 0.016 in. thick. The insulation minimizes heat losses from the building and maintains the required metal temperature during cold weather, and reduces the summer air-conditioning load.

The shell includes two airlocks. The principal access to the shell will be through the personnel airlock that connects the containment building to the turbine building. The airlock is 21 ft. 6 in. long between its two doors, which are 5 ft. 6 in. by 7 ft. and are large enough to permit passage of a spent fuel element shipping cask. The containment building can also be evacuated, if necessary, through the emergency airlock, which is 7 ft. long and 5 ft. in diameter, with two circular doors of 32-1/2 in. diameter (with a 30-in. opening). Both airlocks are at el. 642 ft. 9 in. and lead to platform structures from which descent to grade level can be made. When the doors are closed, a clamp exerts a positive force, which is transmitted through the doors to live-rubber gaskets around the door frames to ensure gas tightness.

An 8 ft. by 10 ft. freight door opening in the containment building accommodates large pieces of equipment. It will be opened only when containment isolation is not required. Nine-inch-thick concrete blocks are placed on the outside of the door for shielding. The door is bolted

internally to the door frame in the shell. Two rubber gaskets between the door and doorframe ensure a pressure-tight seal.

Approximately 300 mineral insulted (MI) cables and 75 bulkhead conductors penetrate the containment shell. These are in the northwest quadrant of the shell adjacent to the electrical room under the control room. The majority of pipe penetrations leave the containment vessel 1 to 10 ft. below grade level and enter either at the northwest quadrant into the pipe tunnel that runs to the turbine building, or on the northeast side into the tunnel connecting the turbine building, reactor building, stack, and the waste treatment and waste gas storage areas.

An approximately 45,000-gal. storage tank in the dome of the containment building supplied water for the emergency core spray system and the building spray system. The piping connection to the emergency core spray system is near the bottom of the tank. The connection to the spray headers of the building spray system is a standpipe within the tank; the top of the standpipe is sufficiently above the bottom of the tank to leave 15,000 gal. of water for use in the emergency core spray system. The storage tank also provides water for use during refueling and other operations in the fuel element storage well.

A 50-ton traveling bridge crane with a 5-ton auxiliary hoist is located in the upper part of the containment building. The bridge completely spans the building and travels on circular tracks supported by columns around the inside of the building just below the hemispherical upper head. A trolley containing all the lifting mechanisms travels on the bridge to near the crane rail, and it permits crane access to any position on the main floor under the trolley travel-diameter. The lifting cables of both the 50-ton and the 5-ton hoists are also long enough to reach down through hatchways into the basement area. Hatches at several positions in the main and intermediate floors may be opened to allow passage of the cables and equipment.

4.2.2 Turbine Building

The general location of the Reactor and Turbine Buildings is shown in Figure 4.3. The Turbine Building contains a major part of the power plant equipment. The turbine-generator is on the main floor. Other equipment is located below the main floor. This equipment includes the feedwater heaters, reactor feedwater pumps, air ejector, vacuum pump, full flow demineralizers, offgas recombiner and cooler, condensate pumps, air compressors, air dryer, oil purifier, service water pumps, component cooling water coolers and pumps, make-up water demineralizer system, domestic water heater, turbine oil reservoir, oil tanks and pumps, turbine condenser, unit auxiliary transformer, 2400-volt and 480-volt switchgear, motor control centers, diesel engine-generator sets, emergency storage batteries, inverters and other electrical, pneumatic, mechanical and hydraulic systems and equipment required for a complete power plant. A 30/5-ton capacity, pendant-operated overhead electrical traveling crane spans the Turbine Building. The crane has access to major equipment items located below the floor through numerous

hatches in the main floor. A 40-ton capacity, pendant-operated overhead electric crane spans the space between turbine building loading dock and Waste Treatment Building.

The Turbine Building also contains the main offices, the Control Room (for both turbine-generator and reactor), locker room facilities, laboratory, shops, counting room, personnel change room, and decontamination facilities, heating, ventilating and air conditioning equipment, rest rooms, storeroom, and space for other plant services. In general, these areas are separated from power plant equipment spaces. The Control Room is on the main floor on the side of the Turbine Building that is adjacent to the Containment Building. The general arrangement of the Containment and Turbine Buildings is shown in Figures 4.3 through 4.5.

4.2.3 Waste Treatment Building and LSA Storage Building

The Waste Treatment Building (WTB) is located to the northeast of the Containment Building. The building contains facilities and equipment for decontamination and the collection, processing, storage, and disposal of low level solid radioactive waste materials in accordance with the Process Control Program.

The grade floor of the Waste Treatment Building contains a shielded compartment which encloses a 320 ft^3 stainless steel spent resin receiving tank with associated resin receiving and transfer equipment. A high integrity disposal liner is located in the adjacent shielded cubicle.

Adjacent to these shielded resin handling cubicles are two open cubicles, one of which is about 3' above grade. The grade level area contains two backwashable radioactive liquid waste filters, the spent resin liner level indication panel and the spent resin liner final dewatering piping, container, and pumps. The second above-grade area is a decontamination facility, consisting of a steam cleaning booth, ultrasonic cleaning bath, a decontamination sink, and heating/ventilation/air conditioning units.

The remaining grade or above grade areas contain a shower/wash/frisking area, and the dry active waste (DAW) compactor unit and temporary storage space for processed DAW containers.

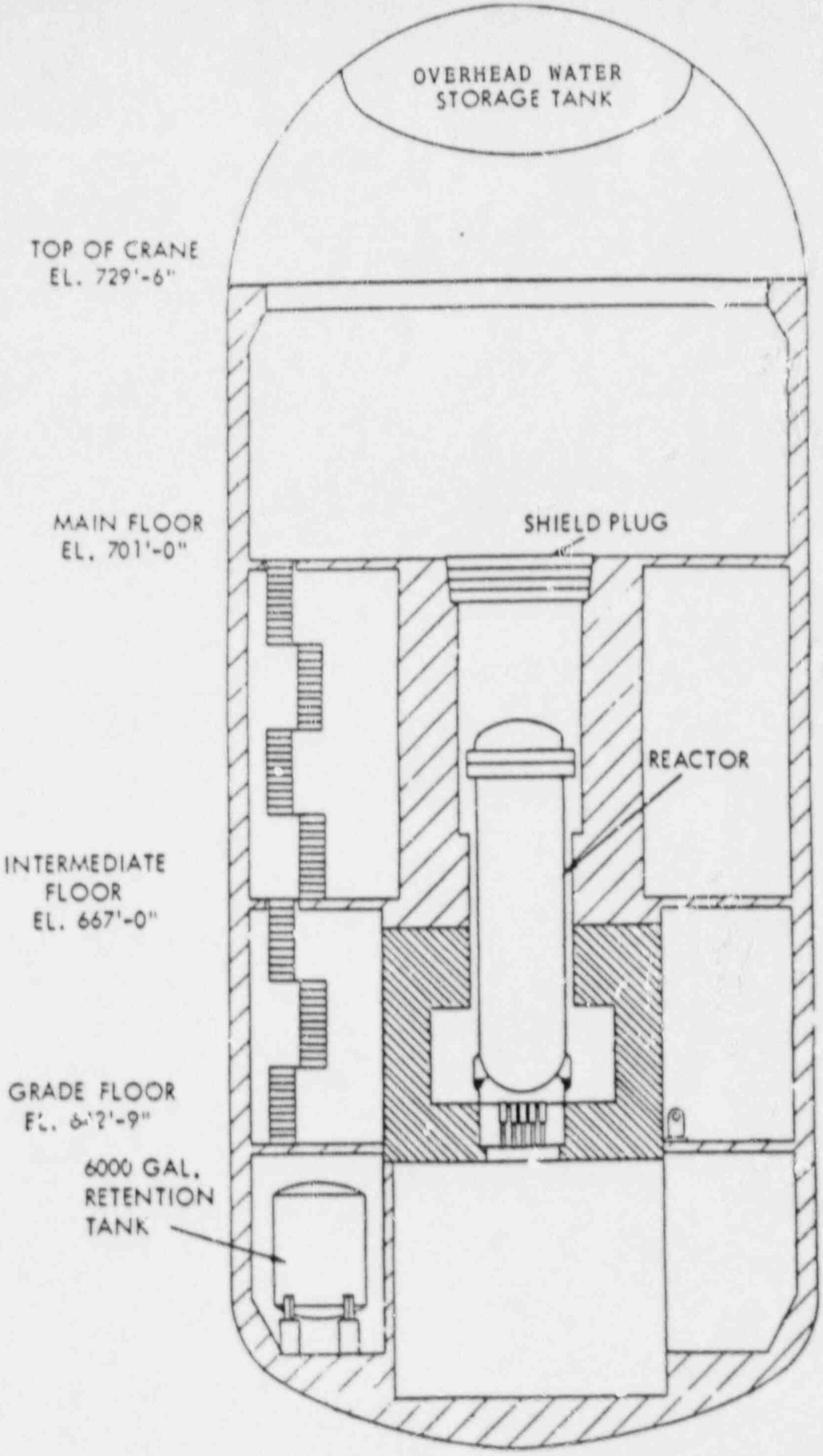
Beneath the grade floor are two shielded cubicles. One cubicle, through which access is gained by removal of floor shield plugs, is available for the storage of up to nine higher activity solid waste drums. The other area, through which access is gained by a stairway, contains the dewatering ion exchanger, the WTB sump and pump, final offgas line HEPA/charcoal filters, and additional waste storage space.

The Waste Treatment Bulding ventilation is routed through an HEPA filter to the stack plenum. The building is normally maintained at a negative pressure. The general arrangement of the WTB is shown on Figure 4.5.

The LSA Storage Building is southwest of the Turbine Building. It is used to store processed, packaged and sealed low level dry active waste materials, and sealed low level activity components for a period of approximately 5 years. The building has the capacity for 500 DOT17H-55 gallon drums of waste. No liquids are stored in this building. There are no effluent releases from this building during normal use.

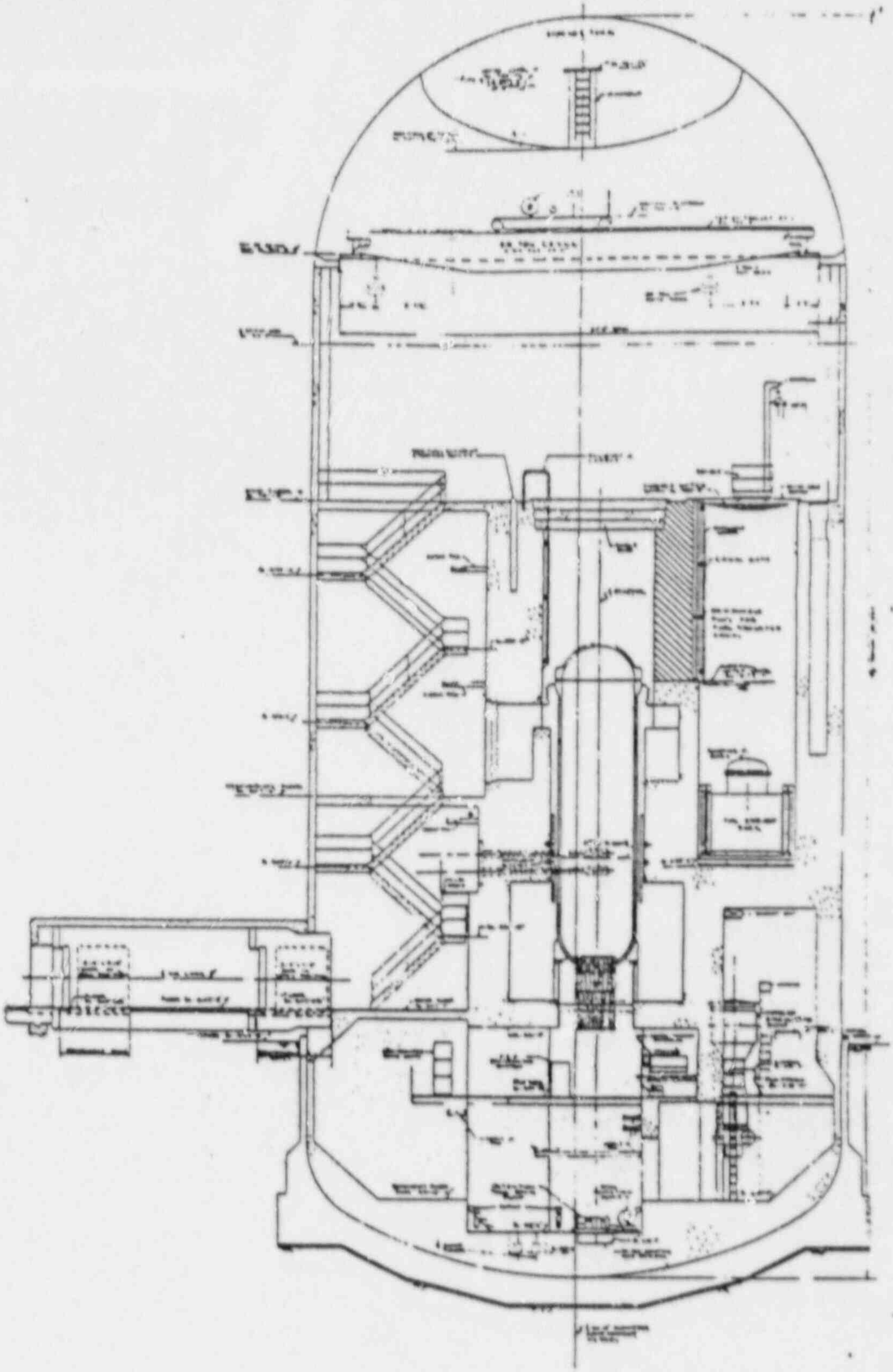
4.2.4 Cribhouse

The Cribhouse is located on the bank of the Mississippi River to the west of the plant and through its intake structure, provides the source of river water to the various pumps supplying river water to the plant. The Cribhouse contains the diesel-driven high pressure service water pumps, travelling screens, low pressure service water pumps and the circulating water pumps.



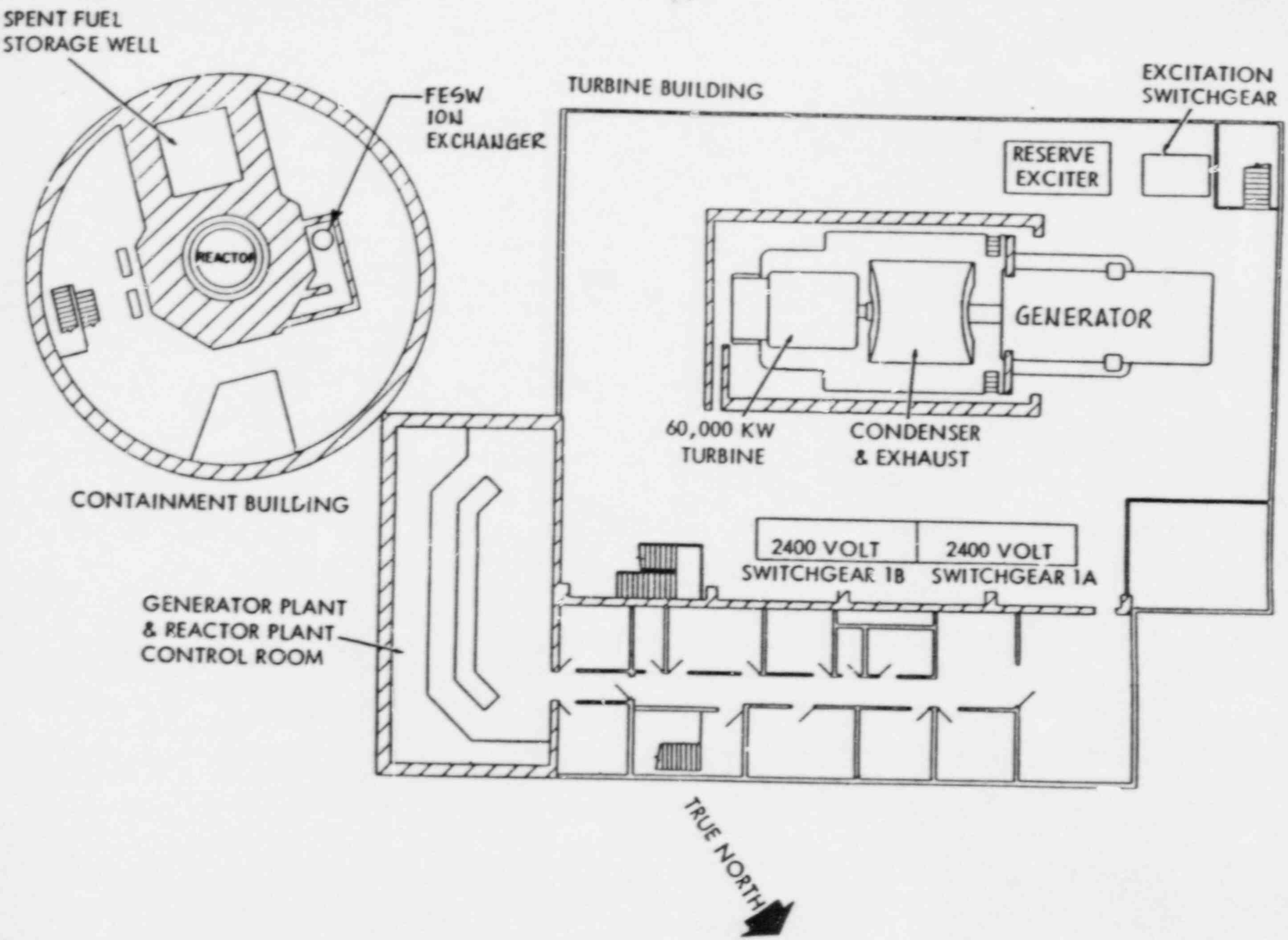
Containment Building Elevation

FIGURE 4.1



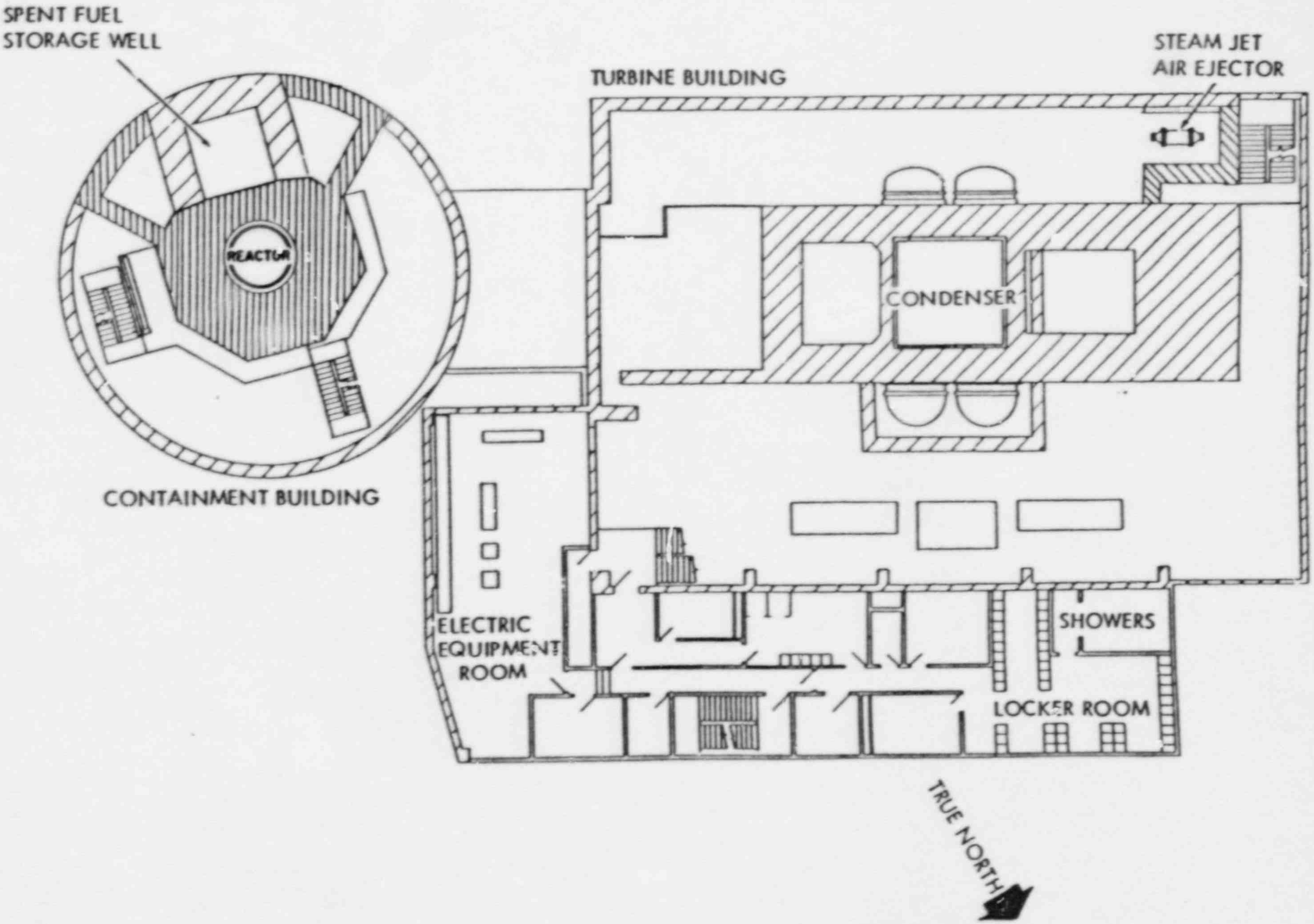
Containment Building General Arrangement

FIGURE 4.2



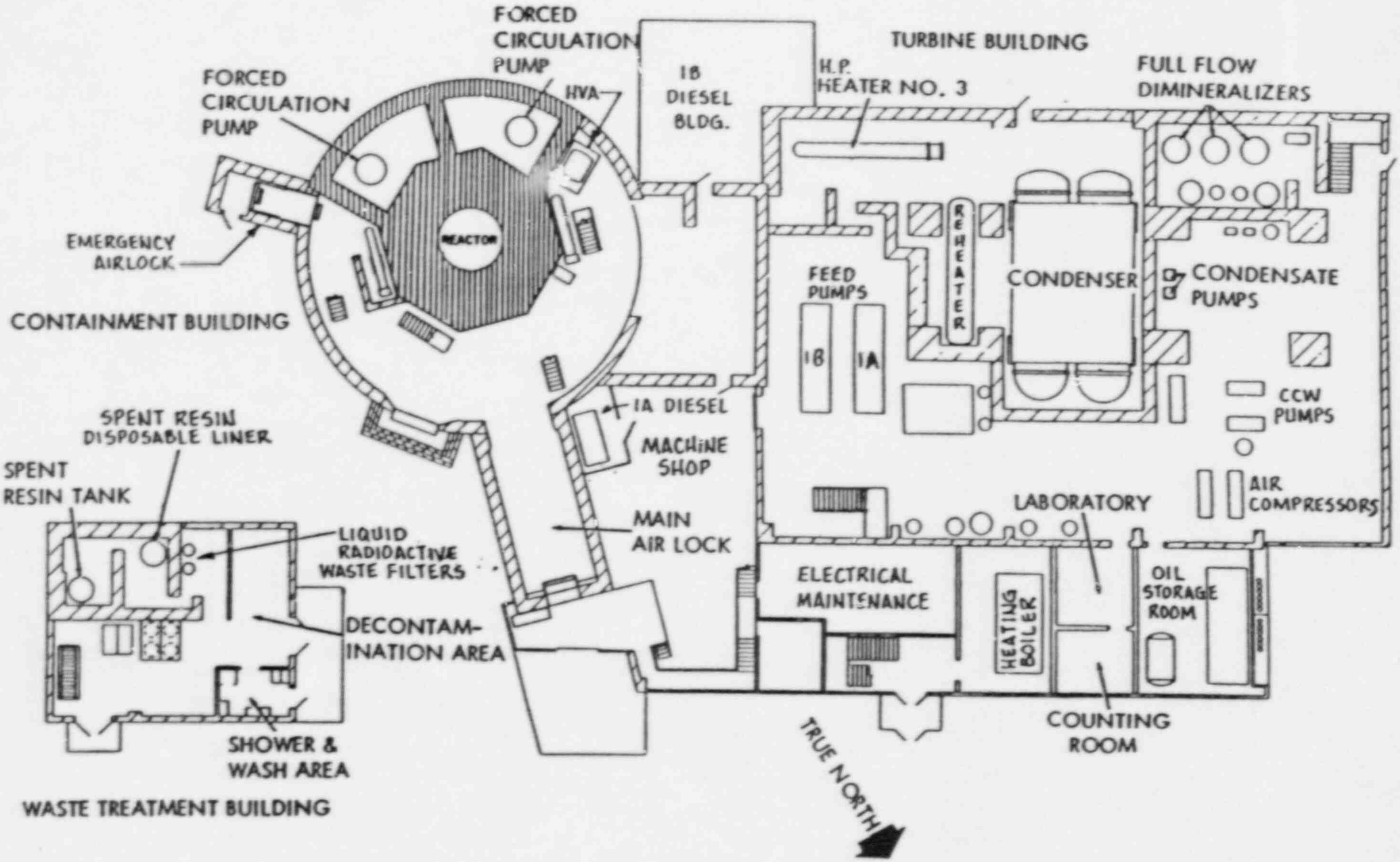
Main Floor of Turbine and Containment Buildings, El. 668'0"

FIGURE 4.3



Mezzanine Floor of Turbine and Containment Buildings, El. 654'0"

FIGURE 1.4



Grade Floor of Turbine, Containment, and Waste Treatment Buildings, El. 640'0"

FIGURE 4.5

5. PLANT STATUS

5.1 FUEL INVENTORY

5.1.1 Spent Fuel

During June 1987 all fuel assemblies were removed from the reactor vessel. Currently there are 333 spent fuel assemblies stored in the spent fuel pool.

This spent fuel consists of three different types of fuel assemblies. Type I (82 assemblies) and Type II (73 assemblies) were fabricated by Allis-Chalmers (A-C) and Type III (178 assemblies) by EXXON. All of the fuel assemblies are 10x10 arrays of Type 348 stainless steel clad rods with stainless steel and Inconel spacers and fittings. The initial enrichment of the uranium in the Type I and Type II fuel was 3.63% and 3.92% respectively and the nominal average initial enrichment of the Type III fuel was 3.69%. The Type III assemblies contain 96 fueled rods and 4 inert Zircaloy-filled rods.

The 72 fuel assemblies removed from the reactor in June 1987 have assembly average exposures ranging from 4,678 to 19,259 megawatt-days per metric ton of uranium. The exposures of the 261 fuel assemblies discharged during previous refuelings range from 7,575 to 21,532 MWD/MTU. The oldest fuel stored was discharged from the reactor in August 1972. Forty-nine of the A-C fuel assemblies discharged prior to May 1982 contain one or more fuel rods with visible cladding defects and 54 additional A-C fuel assemblies discharged prior to December 1980 contain one or more leaking fuel rods as indicated by higher than normal fission product activity observed during dry sipping tests.

The estimated radioactivity inventory in the 333 spent fuel assemblies is tabulated in Table 5-1.

5.1.2 New Fuel

Fifty-two new, unused, EXXON fuel assemblies are presently stored in the locked new fuel storage cabinets located in the containment building in the reactor biological shield. This new fuel will be shipped offsite within a few months for recovery of the contained material.

TABLE 5-1

SPENT FUEL RADIOACTIVITY INVENTORY

January 1988[a]

Radio-nuclide	Half Life (Years)[b]	Activity (Curies)	Radio-nuclide	Half Life (Years)[b]	(Curies)
^{144}Ce	7.801 E-1	2.636 E+6	^{90}Sr	2.770 E+1	1.147 E+6
^{137}Cs	3.014 E+1	1.666 E+6	^{241}Pu	1.440 E+1	1.138 E+6
^{106}Ru	1.008 E+0	1.524 E+6	^{55}Fe *	2.700 E+0	5.254 E+5

TABLE 5-1 - (cont'd)

Radio-nuclide	Half Life (Years)[b]	Activity (Curies)	Radio-nuclide	Half Life (Years)[b]	(Curies)
^{95}Zr(Nb)	1.754E-1(9.58E-2)	3.555 E+5	^{95}Zr *	1.750 E-1	3.52 E+2
^{134}Cs	2.070 E+0	3.291 E+5	^{59}Ni *	8.000 E+4	2.87 E+2
^{85}Kr	1.072 E+1	1.160 E+5	^{99}Tc	2.120 E+5	2.76 E+2
^{110m}Ag	6.990 E-1	1.018 E+5	^{125}Sb	2.760 E+0	2.73 E+2
^{89}Sr	1.385 E-1	1.009 E+5	^{155}Eu	4.960 E+0	1.68 E+2
^{127m}Te	2.990 E-1	8.238 E+4	^{234}U	2.440 E+5	6.37 E+1
^{60}Co *	5.270 E+0	6.395 E+4	^{243}Am	7.380 E+3	6.31 E+1
^{103}Ru	1.075 E-1	6.334 E+4	^{113m}Cd	1.359 E+1	1.78 E+1
^{147}Pm	2.620 E+0	4.129 E+4	^{94}Nb *	2.000 E+4	1.59 E+1
^{63}Ni *	1.000 E+2	3.540 E+4	^{135}Cs	3.000 E+6	1.40 E+1
^{141}Ce	8.890 E-2	2.638 E+4	^{238}U	4.470 E+9	1.22 E+1
^{242}Cm	4.459 E-1	1.858 E+4	^{156}Eu	4.160 E-2	8.63 E+0
^{241}Am	4.329 E+2	1.474 E+4	^{242}Pu	3.760 E+5	8.58 E+0
^{238}Pu	8.774 E+1	1.262 E+4	^{236}U	2.340 E+7	6.32 E+0
^{239}Pu	2.410 E+4	8.837 E+3	^{121m}Sn	7.600 E+1	4.44 E+0
^{240}Pu	6.550 E+3	7.165 E+3	^{237}Np	2.140 E+6	2.19 E+0
^{154}Eu	8.750 E+0	4.020 E+3	^{235}U	7.040 E+8	1.89 E+0
^{244}Cm	1.812 E+1	3.603 E+3	^{151}Sm	9.316 E+1	1.51 E+0
^{51}Cr *	7.590 E-2	3.002 E+3	^{126}Sn	1.000 E+5	7.01 E-1
^{129m}Te	9.340 E-2	1.170 E+3	^{79}Se	6.500 E+4	5.52 E-1
^{3}H	1.226 E+1	5.510 E+2	^{129}I	1.570 E+7	3.90 E-1
^{59}Fe *	1.220 E-1	5.120 E+2	^{93}Zr	1.500 E+6	1.11 E-1
^{152}Eu	1.360 E+1	5.110 E+2	^{131}I	2.200 E-2	2.00 E-3
^{242m}Am	1.505 E+2	4.900 E+2			

(a) Computer Program, FACT-1, DPC, July 1987, and hand calculations.

(b) Computer Program, TPASGAM, Nuclide Identification Package, J. Keller, Analytical Chemistry Division, ORNL, June 1986.

* Activity in fuel assembly hardware based on neutron activation analysis.

5.2 PLANT SYSTEMS AND THEIR STATUS

5.2.1 Reactor Vessel and Internals

The reactor vessel consists of a cylindrical shell section with a formed integral hemispherical bottom head and a removable hemispherical top head which is bolted to a mating flange on the vessel shell to provide for vessel closure. The vessel has an overall inside height of 37 feet, an inside diameter of 99 inches, and a nominal wall thickness of 4 inches (including 3/16-inch of integrally bonded stainless steel cladding).

The reactor vessel is a ferritic steel (ASTM A-302-Gr-B) plate with integrally bonded Type 304L stainless steel cladding. The flanges and large nozzles are ferritic steel (ASTM A-336) forgings. The small nozzles are made of Inconel pipe.

The reactor internals consist of the following: a thermal shield, a core support skirt, a plenum separator plate, a bottom grid assembly, steam separators, a thermal shock shield, a baffle plate structure with a peripheral lip, a steam dryer with support structure, an emergency core spray tube bundle structure combined with fuel holddown mechanism, control rods and the reactor core.

System Status

The reactor core (fuel elements and startup source) has been removed. The 29 control rods remain in the reactor vessel. The reactor vessel head is installed and partially bolted in place. The reactor vessel is maintained in wet layup. The vessel and primary systems may eventually be placed in dry layup. The control rods may be removed to the Fuel Element Storage Well or a licensed facility during SAFSTOR.

5.2.2 Forced Circulation System

The Forced Circulation System was designed to circulate sufficient water through the reactor to cool the core and to control reactor power from 60 to 100 percent.

Primary water passes upward through the core, and then down through the steam separators to the recirculating water outlet plenum. The water then flows to the 16-in. forced circulation pump suction manifold through four 16-in. nozzles and is mixed with reactor feedwater that enters the manifold through four 4-in. connections. From the manifold, the water flows through 20-in. suction lines to the two 15,000 gpm variable-speed forced-circulation pumps. The pumps are above the basement floor, within their own shielded cubicles. Hydraulically-operated rotoport valves are at the suction and discharge of each pump. The 20-in. pump discharge lines return the water to the 16-in. forced-circulation pump discharge manifold. From the manifold, the water flows through four equally spaced 16-in. reactor inlet nozzles to the annular inlet plenum, and then downward along the bottom vessel head to the core inlet plenum.

The system piping is designed for a maximum working pressure of 1450 psig at 650°F (a pressure above the maximum reactor working pressure to allow for the static head and the pump head).

Since the piping from the reactor to the rotoport valves is within the biological shield and is not accessible, the valves and piping are clad with stainless steel. The piping between the rotoport valves and the pumps is low-alloy steel. Provisions have been made for determining the rate and type of any corrosion, and the low-alloy piping can be replaced if the corrosion rate is excessive. To facilitate repair or replacement, decontamination solutions can be circulated to remove radioactive particles.

Each forced circulation pump has an auxiliary oil system and a hydraulic coupling oil system. Each auxiliary oil system supplies oil to cool and lubricate the three (1 radial and 2 thrust) pump coupling bearings. Each hydraulic coupling oil system supplies cooled oil at a constant flow rate to the hydraulic coupling.

System Status

The forced circulation system is in wet layup with the capability of being operated for flushing operations. This system may eventually be placed in dry layup.

5.2.3 Seal Injection System

The Seal Injection System provided cooling and sealing water for the seals on the two Forced Circulation (F.C.) Pumps and the 29 control rod drive units.

The Seal Injection System has two positive-displacement pumps, supplied with water from the seal injection reservoir. The reservoir is supplied from the Condensate Demineralizer System with a backup supply from the overhead storage tank. One pump is required for operation with the other on standby.

A cartridge-type filter is provided in the pump suction header. A deaerator, vented to the seal injection reservoir, is located on the suction of the pumps to remove entrained air from the system in the event air is introduced during makeup to the system.

Bladder-type accumulators are connected to the suction and discharge of each seal injection pump. The suction accumulators reduce pump and piping vibrations. The discharge accumulators dampen the pulsating flow from the pumps to provide the constant flow rate which is required for stable system conditions.

The water to the F.C. pump seals passes through a full-flow filter. There are two filters arranged in parallel with one normally in service and the other in standby. The filtered water is then distributed to the two F.C. pump seals. Each seal supply line contains a flow control valve, check valve, and a three-way valve for switching injection points on the seal. Both valves in each supply line can be operated from the Control Room.

The water to the control rod drive seals is filtered in the same manner as that of the F.C. pumps seals. The water passes through a flow control valve which maintains a constant flow rate to the seals. The individual supply lines to the 29 control rod drive units each have a throttle valve and flow indicator for setting the required flow rate to each seal. The normal leakoff from the seals (0.15 gpm or less) is drained to the reactor basement floor sump.

Continuous blowdown of water from the control rod drive upper housing is removed through a connection on each control rod drive upper housing flange. Each line contains a throttle valve, and all lines join at a common manifold, from which suction is taken by the control rod nozzle effluent pumps. The pumps discharge into the inlet of the Decay Heat System, which is directly connected to the Forced Circulation System.

System Status

This system is still capable of operating for operation of the Forced Circulation Pumps during flushing operations. This system may eventually be placed in dry layup.

5.2.4 Decay Heat Cooling System

The Decay Heat Cooling System is a single high pressure closed loop containing a pump, cooler, interconnecting piping, and the necessary instrumentation.

The decay heat loop is connected across the 20-in. Forced Circulation Pump 1A supply and return lines on the reactor side of the Forced Circulation Pump isolation valves. Both connections to the forced circulation piping are located outside the reactor biological shield.

Reactor water from the Forced Circulation Pump supply line passes through an 8-in. line to the suction side of the Decay Heat Pump. The water is then discharged through a 6-in. line into the Decay Heat Cooler where it is cooled and returned to the Forced Circulation Pump return line.

A 2-in. blowdown line to the Main Condenser takes off downstream of the Decay Heat Cooler. This may be used to remove the excess water due to seal inleakage and thermal expansion of the water.

Another 2-in. line downstream of the Decay Heat Cooler connects to the reactor vessel head vent line which can be used to promote better circulation of the water within the reactor vessel head.

System Status

This system is not required to be operable; however, the 2-in. blowdown line to the Main Condenser may be used during flushing operations. This system may eventually be placed in dry layup.

5.2.5 Emergency Core Spray System

The Emergency Core Spray System consists of a spray header with individual spray lines for each fuel assembly mounted inside the reactor vessel and two positive displacement-type pumps which take suction from the Overhead Storage Tank or the High Pressure Service Water (HPSW) header.

The two Emergency Core Spray Pumps are positive displacement-type pumps rated for 50 gpm at a design pressure of 1450 psig. Each pump is driven by a 50-hp, 440-volt, 3-phase motor suitable for operation under MCA conditions. The motors are supplied from Essential Bus 1A and Essential Bus 1B, respectively.

A low pressure supply line with a control valve bypasses the Core Spray Pumps to allow the demineralized water from the Overhead Storage Tank, or the service water from the HPSW supply line, to flow directly to the core spray header. The control valve switch on Bench E must be in the "AUTO" position and a reactor water low level (-12 inches) and reactor low pressure (<30 psig) conditions must be met before the valve will open. The flow from the Overhead Storage Tank to the spray nozzles has been calculated to be approximately 85 gpm assuming that the reactor vessel and the Containment Building are at the same pressure.

The core spray line penetrates the north wall of the biological shield at approximately 11 feet above the intermediate floor and enters the reactor vessel through a 1-1/2-inch nozzle on the northwest quadrant. The core spray header above the top of the core spray tube support grid, supplies the 72 spray lines. An individual 3/8-inch spray line is provided for each fuel element.

The spray lines are installed concentrically within tubes. Each spray line contains a needle valve on the spray header used to set the flow for each fuel assembly location. The valve stems are staked after being set so the required flow will be obtained when the reactor is operating at 577°F. The required flow per assembly varies between 0.40 and 0.87 gpm, depending on the assembly location.

System Status

Since the reactor is defueled, the High Pressure Core Spray System is not required to be operable. However, due to wording in the Technical Specifications, the Low Pressure Core Spray Mode, control valve 53-25-001, is still required to be operable. When the required Technical Specification change is approved, the system will be removed from operable status.

5.2.6 Boron Injection System

The Boron Injection System consists of a Boron Tank and the necessary valves and piping to connect the tank to the emergency core spray pumps suction header, and to connect the pumps discharge header to the boron injection nozzle on the forced-circulation discharge header.

The tank normally contained approximately 800 gallons of a nominal 17.8 w/o solution of sodium pentaborate decahydrate. During the first minute of injection, approximately 70 gallons of the 17.8 w/o solution would be injected in the reactor, which after dilution with the primary water, provides sufficient negative reactivity to overcome the voids and Doppler effect and a 0.4 percent reactivity safety margin.

System Status

Since the reactor is defueled, this system is not required to be operable.

5.2.7 Primary Purification System

The Primary Purification System is a high pressure, closed loop system consisting of a Regenerative Cooler, Purification Cooler, Pump, two Ion Exchangers and filters.

The functions of the Primary Purification System are:

(1) to maintain optimum reactor water quality (pH and conductivity) to minimize corrosion,

(2) to remove dissolved and suspended solids in order to minimize fouling of heat transfer surfaces, pipes and vessels, and maintain primary coolant activity level low, and

(3) to provide an alternate means of removing excess reactor water.

System Status

This system is presently in operation to clean up the primary system and maintain water quality. This system may eventually be placed in dry layup.

5.2.8 Alternate Core Spray System

The Alternate Core Spray System consists of two diesel-driven High Pressure Service Water (HPSW) pumps which take a suction from the river and discharge to the reactor vessel through duplex strainers and two motor-operated valves installed in parallel.

The Alternate Core Spray System was installed to provide backup for the High Pressure Core Spray System. It provided further assurance that melting of fuel-element cladding will not occur following a major recirculation line rupture. It has a secondary function of providing backup to the High Pressure Service Water System and Fire Suppression System.

The Emergency Service Water Supply System (ESWSS) Pumps were portable pumps which served as backups to the diesel-driven High Pressure Service Water Pumps in the event the Cribhouse or underground piping were damaged.

System Status

Since the reactor is defueled, the Alternate Core Spray System is not required to be operational. Therefore, the manual isolation valve to the Containment Building is closed. The ESWSS pumps are not required to be operable. The balance of the system continues to be operational to provide the requirements of the HPSW System.

5.2.9 Control Rod Drive Auxiliaries

The function of the Control Rod Drive Nitrogen System is to charge each of the 29 control rod drive scram accumulators with nitrogen.

The function of the Control Rod Drive Hydraulic Charging System is to charge each of the 29 control rod drive scram accumulators with hydraulic fluid.

The Control Rod Drive Nitrogen System consists of: nitrogen booster compressor, high pressure nitrogen bottles, low pressure nitrogen bottles and associated manifolds, valves, instruments and controls.

The system supplies nitrogen at pressures up to 3000 psig to each of the 29 scram accumulators by use of the high pressure manifold through a variable pressure reducer. A pressure gauge located adjacent to and downstream from the manual charging valve gives visual indication of the pressure during charging.

Pressure in the high pressure nitrogen bottles is maintained at approximately 6000 psig by the nitrogen booster compressor which takes its suction from 2500 psi nitrogen bottles.

The Hydraulic Charging System is a packaged unit consisting of a 60-gallon oil reservoir, two positive-displacement pumps, filters, valves, a surge accumulator, a charging manifold, tubing, and the necessary instrumentation and controls.

Each charging pump takes suction from the oil reservoir through a sump strainer and discharges the oil at high pressures through two 10-micron filters, a stop valve, a check valve, and into a discharge line common to both pumps. A surge accumulator is installed in the discharge line upstream from the charging manifold to dampen the pressure fluctuations from the hydraulic charging pumps. The high-pressure oil is distributed at the charging manifold to each of the 29 scram accumulators.

System Status

Neither the Control Rod Drive Auxiliaries nor the Control Rod Drives are required to be operational.

5. PLANT STATUS - (cont'd)

5.2.10 Gaseous Waste Disposal System

A 4-inch offgas header routed the gases from the Main Condenser Air Ejectors and the shutdown condenser to the 150 ft^3 holdup tank. One of two air dryers were used to dehumidify the offgas prior to entering the holdup tank. The holdup tank provided approximately 10 minutes holdup time, which permitted decay of short-lived radioactive isotopes. From the holdup tank the gas passed through one of two HEPA and charcoal filter units with charcoal granules (for iodine absorption) and was monitored for gaseous and particulate activity. From the filters the waste gas could be routed directly to the base of the stack, or to the gas storage tanks, utilizing a steam ejector to draw the gases from the holdup tank. The gas mixture then proceeded to the recombiner where the hydrogen and oxygen in the waste gas were recombined, thus reducing the storage load of the waste gas system and reducing the possibility of a combustible gas mixture in the system.

The steam in the gas was condensed in the recombiner condenser, and by vacuum, pulled to main condenser. The gas was routed to the two 1600 ft^3 gas storage tanks.

Normally the gas storage tanks were lined up for series offgas flow. Offgas entered the inlet of the "B" tank, then flowed to the "A" tank. The gas stream traversed the entire length of each tank. This added approximately 10.5 hours of holdup time. The offgas was then routed to the base of the stack through a remotely-operated control valve and one of two HEPA and charcoal filter units with charcoal granules (for iodine adsorption).

High radiation on the offgas activity monitor after the holdup tank automatically shifted the offgas routing to tanks (if routed to stack) and closed the offgas storage tanks discharge valve.

System Status

This system is no longer used.

5.2.11 Fuel Element Storage Well System

The storage well is a stainless lined concrete structure 11 feet by 11 feet by approximately 42 feet deep. When full, it contains approximately 38,000 gallons.

It is completely lined with Type 316 stainless steel. The walls are 16-gauge sheet and the bottom a 3/8-inch plate. All joints are full penetration welds. Vertical and horizontal expansion joints in the storage well allow for thermal expansion.

Design values for the storage well are given below:

Well Floor: safe uniform live load . . . . 5,000 lb/ft^2

Spent fuel elements and control rods are stored in two-tiered racks in the Fuel Element Storage Well until they can be shipped. A transfer canal connects the upper portion of the well to the upper vessel cavity and is closed with a water-tight gate and a concrete shield plug. The water level in the well is normally maintained at an elevation of $\geq$ 695 feet with fuel in upper rack.

Storage well cooling is accomplished by drawing water through a 6-inch penetration at elevation 679 feet, or a 4-inch line at elevation 679 feet 11 inches, and pumping it through the fuel storage well cooler and returning it at the bottom of the well, with either of two storage well pumps.

Cleanup is provided by the FESW ion exchanger. A 4-inch line from the Overhead Storage Tank is used to flood the well or pump water back to the Overhead Storage Tank. Overflow and drain pipes from the well and cavity are routed to the retention tanks.

Normal makeup to the storage well is provided by demineralized water through the "FESW Remote Operated Fill Valve," which is operated from Benchboard E in the Control Room.

The cooling system is conservatively designed to remove the decay heat of a full core one week after shutdown, with the storage well water at 120°F and the ultimate heat sink, the river, at 85°F.

System Status

The Fuel Storage Well contains 333 irradiated fuel elements, 10 control rods, startup sources and a number of zirconium and stainless steel shroud cans. The Fuel Element Storage Well System will remain in operation as part of the SAFSTOR Program until all fuel is sent offsite.

5.2.12 Component Cooling Water System

The Component Cooling Water System provides controlled quality cooling water to the various heat exchangers and pumps in the Reactor Building. It also serves as an additional barrier between radioactive systems and the river.

The Component Cooling Water System is a closed system consisting of two pumps, two heat exchangers, a surge tank, and the necessary piping, valves, controls, and instrumentation to distribute the cooling water.

The Component Cooling Water Pumps, Coolers, and the Surge Tank are located in the Turbine Building. Water flows from the pumps, to the cooler, and then to the component cooling water supply header in the Reactor Building.

The flow requirements of the components cooled by the Component Cooling Water System were as follows during plant operation:

		Design (GPM)	Nominal (GPM)
(1)	FCP Hydraulic Coupling Coolers	60	60
(2)	FCP Lube Oil Coolers	30	30
(3)	Shield Cooler	75	75
(4)	Control Rod Nozzle Effluent Pumps	30	30
(5)	Purification Pump	15	15
(6)	Purification Cooler	260	200
(7)	Reactor Building Air Conditioners	60 ea	120
(8)	Decay Heat Pump	20	20
(9)	Decay Heat Cooler	570	100
(10)	Fuel Element Storage Well Cooler	260	100
(11)	Sample Coolers	5-10 ea	40
(12)	Failed Fuel Element Location System Cooler	40	0
(13)	Station Air Compressors	20 ea	20
(14)	PASS		
	(a) Reactor Coolant Sample	10	5
	(b) Containment Atmosphere Sample	40	40
	TOTAL	1560	855

Water from each of the components, listed above, flows to the component cooling water return header. This header leaves the Reactor Building and connects to the suction of the Component Cooling Water Pumps. A sample stream from the supply header is monitored for radioactivity and returned to the suction header. The temperature of the water in the supply header is automatically controlled by varying the Low Pressure Service Water flow to the tube side of the Component Cooling Water Coolers.

System Status

This system remains in operation to provide cooling water to the Fuel Element Storage Well Cooler, Air-Conditioners and the Station Air Compressors.

5.2.13 Shield Cooling System

The Shield Cooling System was designed to maintain the temperature of the thermal shield and biological shield concrete below 140°F and 150°F, respectively. The system is a closed loop containing 26 stainless steel cooling coils, a cooler, two pumps, a filter, a surge tank, piping, valves, and the necessary instrumentation. Two centrifugal pumps (one operating, one standby) circulate 75 gpm of treated demineralized water through the closed loop. The coolant from the pump passes through a full-flow filter and into the cooling coil supply header. From the supply header, the coolant passes through the 26 individual cooling coils and is returned to a common return header. The coolant from the return header passes through the cooler, and returns to the suction side of the pump. The cooler transfers the heat from the coolant to the Component Cooling Water System. A surge tank connected to the return line upstream of the cooler, acts as a head tank and accommodates any water expansion and contractions due to temperature changes.

System Status

This system is not required to be operational.

5.2.14 Shutdown Condenser System

The primary function of the Shutdown Condenser was to provide a backup heat sink for the reactor, in the event the reactor was isolated from the main condenser, by the closure of either the Reactor Building Steam Isolation valve or the Turbine Building Steam Isolation valve. In addition, the Shutdown Condenser acted as an over-pressure relief system in limiting over-pressure transients.

The Shutdown Condenser is located on a platform 10 feet above the main floor in the Reactor Building. Steam from the 10-inch main steam line passed through a 6-inch line, two parallel inlet steam control valves, back to a 6-inch line and into the tube side of the condenser where it was condensed by evaporating cooling water on the shell side. The steam generated in the shell was exhausted to the atmosphere through a 14-inch line which penetrates the Reactor Building. An area monitor is located next to the steam vent line near the containment shell penetration in order to detect excessive activity release in the event of Shutdown Condenser tube failures. The main steam condensate was collected in the lower section and returned to the reactor vessel by gravity flow. The condensate line leaving the condenser is a 6-inch line along the horizontal run and is reduced to 4 inches for the vertical section. Two parallel condensate outlet control valves are located in the 4-inch return line. The condensate line also contains two 2-inch vent lines which join together and return to the lower section of the condenser for returning any vapors and/or non-condensible gases which were carried into the condensate line to prevent perturbations in the condensate flow leaving the condenser. The lower section in turn was vented to the offgas system through a 1-inch vent line. Flow in this vent line is restricted by a 1/16th-inch orifice, which is built into and is an integral part of the shutdown condenser offgas control valve seat.

A vent line containing two parallel control valves is connected to the 6-inch condensate return line. The valves discharge directly to the Reactor Building atmosphere and were capable of remote manual operation to vent the primary system directly to the Reactor Building atmosphere under emergency conditions. They performed the function of "Reactor Emergency Flooding Vent Valves" to equalize water level in the building with that in the reactor vessel, for a below-core break, and "Manual Depressurization System (MDS)" to rapidly depressurize the reactor vessel, on failure of the HPCS coincident with a major leak.

System Status

This system is not required to be operational.

5.2.15 Hydraulic Valve Accumulator System

The major components of the Hydraulic Valve Accumulator System are mounted on a common bed plate on the grade floor of the Containment Building. The system consists of a water accumulator tank, a water return sump tank, two air compressors, two water pumps, piping, valves, and the necessary instrumentation and controls.

Approximately 300 gallons of demineralized water is maintained in the Water Accumulator Tank by pumps which take a suction from the Water Return Sump Tank. The level is automatically maintained by a float switch which operates the pumps as required. The water is stored in the Accumulator Tank under 140 psi air pressure which is supplied by air compressors. The air compressors are automatically controlled by a pressure switch and are interlocked with the level float switch to prevent the compressors from running while a pump is running.

The function of the Hydraulic Valve Accumulator System is to supply the necessary hydraulic force to operate the five piston-type valve actuators, which operate the five Rotoport valves in the Forced Circulation and Main Steam Systems.

System Status

This system is presently maintained operational to operate the Forced Circulation Pump 1A and 1B Suction and Discharge Valves, during flushing operations. This system may be removed from service in the future.

5. PLANT STATUS - (cont'd)

5.2.16 Well Water System

Water for this system is supplied from two deep wells. Well No. 4 is located 115 feet southeast of the containment vessel center, and Well No. 3 is located 205 feet northeast of this centerline. The wells are 12 inches in diameter, with 8-inch pump casings and piping. The upper 40 feet of casing is set in concrete. The pumps are sealed submersible pumps. They take suction through stainless steel strainers, and they discharge into pressure tanks.

The system supplies water to the plant and office for sanitary and drinking purposes and to the generator, radwaste and containment washdown stations. Water supplied by the system is used at personnel and material decontamination stations, at five (5) emergency showers, and at three eyewash stations. It is used as cooling water for the two Turbine Building air-conditioning units and in the boiler blowdown flash tank and sample cooler. The well water system is the source of supply to the plant makeup demineralizer system, the LPSW pumps seal water system, priming water for the lube oil purifier and laundry equipment.

System Status

This system is maintained in continuous operation.

5.2.17 Demineralized Water System

The Virgin Water Tank provides the supply to the Demineralized Water Transfer Pumps which distribute demineralized water throughout the plant, including to the Overhead Storage Tank and the Fuel Element Storage Well Makeup in the Containment Building. Water is demineralized in batches and then stored in the Virgin Water Tank.

Domestic water, under pressure from the well pumps, enters the makeup demineralizer through an isolation valve, a ball flow meter, and an integrating flow meter. The water then flows down through the cation resin bed, out the bottom through an outlet header, and enters the top of the decarbonator through a level control valve.

The demineralized water booster pump takes suction out of the bottom of the decarbonator tank and discharges through a flow meter to the anion tank inlet header. The water flows downward through the anion resin bed and out the bottom outlet header to the 2" outlet line. This line splits into 2" lines; one goes to the Virgin Water Tank, and the other to the Condensate Storage Tank.

The Condensate Storage Tank and the Virgin Water Tank are actually two sections of an integral aluminum tank located on the office building roof. The lower section of this tank is the Condensate Storage Tank, and it has a capacity of 19,100 gallons. The upper, virgin-water, section will hold 29,780 gallons. Both tanks have high- and low-level alarm protection, and each tank level is transmitted to and shown on level indicators in the Control Room.

System Status

The Demineralized Water System will remain in service, mainly as a source of water for the Fuel Element Storage Well and the Turbine Building heating boiler.

The Condensate Storage Tank status is covered under the Condensate System, as it provided the makeup supply for that system.

5.2.18 Overhead Storage Tank

The Overhead Storage Tank is located at the top of, and is an integral part of, the Reactor Building.

The Overhead Storage Tank System consists of the approximately 45,000-gallon tank, the tank level instrumentation and controls, and the piping to the first valve of the systems served by the tank.

The Overhead Storage Tank serves as a backup source for the Seal Injection System; as a reservoir for the water used to flood the Fuel Element Storage Well and upper vessel cavity during fuel handling; and as a receiver for blowdown water from the Primary Purification System. It supplied the water for the Emergency Core Spray System and Containment Building Spray.

System Status

The Overhead Storage Tank remains in use, primarily for a source of makeup water to the Fuel Element Storage Well.

5.2.19 Station and Control Air System

There are two 2-stage compressors and a two-unit single-stage backup compressor. The 2-stage compressors consist of essentially three parts: the low-pressure unit, the high-pressure unit and the motor. One compressor is normally running, and the other compressor is in standby. A pressure switch is installed in the air header; this switch is set to start the standby compressor when the air pressure decreases to 80 psi.

After being compressed, the air passes through an after-cooler and an oil separator to cool the air and to remove moisture and oil from the air before permitting it to enter the air receiver. The air receivers act as a volume storage unit for the station, and they control the distribution stations. The air receiver outlet lines join to form a header for supply to the station and the control air systems. Station air is provided to the Cribhouse, where it is piped to near the suction of the Low Pressure Service Water pumps; to the High Pressure Service Water tank to charge the tank; and to the generator and reactor plants at all floor levels, for station usage as needed.

Control air is supplied from the receiver discharge header through a Hankinson aeroescer filter, Deltech filters and Trinity air dryers to the equipment in the generator and reactor plants. Control air goes to the turbine's initial-pressure regulator, turning gear, generator, Reactor Feed Pumps hydraulic couplings, generator gas pressure transmitter, cribhouse control panel, instrument repair room, building heating and ventilating controls, lower reactor cavity purge, test connections, and to the air-operated valves at pressures reduced from 100 psig to operating ranges of 3 to 15 psig.

A backup instrument air compressor is provided to ensure supply of control air to 1A Emergency Diesel Generator Room and to instruments inside the containment vessel.

The backup instrument air compressor consists of two carbon-ring, 2-cylinder compressors mounted on a surge tank and driven by a single motor. The surge tank is piped so that it floats on the instrument air system downstream of a check valve in the instrument air line at a point just before it enters the Containment Building.

System Status

This system is maintained and in continuous operation.

5.2.20 Low Pressure Service Water System

The system is supplied by two 150 hp 2.3 KV vertical turbine pumps located in the cribhouse through a duplex strainer unit. This system supplies the Component Cooling Water Coolers, Turbine Lube Oil Coolers, Generator Hydrogen Coolers, Condenser Vacuum Pump, Reactor Feedwater Pump, and Circulating Water Pump bushings, and is the normal supply to the HPSW System through the motor-driven pump. The LPSW pump seals are provided with clean well water via a reservoir and two pumps.

System Status

This system is maintained in operation to provide HPSW System pressure under normal conditions and for the Component Cooling Water Coolers.

5.2.21 <u>High Pressure Service Water System</u>

The High Pressure Service Water (HPSW) motor-driven pump takes a positive suction from the Low Pressure Service Water System.

During normal operation, with the motor-driven pump in automatic, system pressure is maintained between 75 $\pm$ 5 and 125 $\pm$ 5 psig, and the pump is protected against a low suction pressure by a 35-psig suction pressure trip. System pressure swings are cushioned by the air space in the High Pressure Service Water surge tank.

Backup protection for the system is provided by two diesel-engine-driven auxiliary service water pumps that will maintain system pressure between 60 and 150 psig.

The HPSW pump discharges into a header that divides into two main loops: one loop serves the Turbine Building, Containment Building, Waste Treatment Building, hose stations and sprinklers, and the other loop supplies the outside fire hydrants and the cribhouse sprinklers.

<u>System Status</u>

This system is maintained in operation to provide fire protection.

5.2.22 Circulating Water System

Circulating water is drawn into the Cribhouse intake flume from the river through traveling screens by circulating water pumps 1A and 1B, which are located in separate open suction bays. Each pump discharges into 42-inch pipe; the pipes join a common 60-inch pipe leading to the main condenser in the Turbine Building. At the condenser, the 60-inch pipe branches into two 42-inch pipes feeding the top section of the water boxes. The main condenser is a two-pass divided water box type. Circulating water enters the top section of the condenser tube side and is discharged from the bottom section tube side. The condenser tubes extend the length of the condenser and are fastened at each end to the tube sheets inserted between the water boxes and the shell.

The 42-inch condenser circulating water outlet lines tie into a common 60-inch line which discharges to the seal well from Genoa No. 3 Plant, located approximately 600 feet downstream from the LACBWR Cribhouse.

System Status

This system is maintained operational for periodic use for dilution of liquid waste discharges.

5.2.23 Condensate System and Feedwater Heaters

The Condensate System took condensed steam from the condenser hotwell and delivered it under pressure to the suction of the reactor feed pumps. Two identical full-capacity condensate pumps took suction from the hotwell, and pumped the condensate through a full-flow demineralizing system, the air ejector condensers, the gland steam exhaust condenser and two feedwater heaters before entering the feed pumps.

The Condensate System also supplied the turbine exhaust sprays, the reactor feed pump shaft sealing cooling system, the normal makeup to the seal injection system, and gland seal steam generator. Hotwell level is maintained by automatic makeup from, or overflow to, the Condensate Storage Tank.

System Status

This system will be flushed to reduce radiation levels and then drained. The Condensate Storage Tank has been left dry.

5.2.24 Reactor Feedwater Pumps

The feedwater pumps took preheated condensate from No. 2 feedwater heater and delivered it through No. 3 feedwater heater to the reactor. The pumps boosted the system pressure from about 200 psi to approximately 1300 psi. The pump coupling arrangement is such that pump speed, and therefore capacity, may be varied to control reactor water level. Each pump is a separate unit containing all the auxiliaries, controls, and other components necessary for independent operation.

System Status

The Reactor Feedwater Pumps are not used.

5.2.25 Full-Flow Condensate Demineralizer System

The Full-Flow Condensate Demineralizer System consists of three service tanks, each with one-half system capacity and arranged in parallel. Its purpose was to remove ionic impurities from the condensate system water before admitting it to the reactor. Each service tank is capable of delivering 700 gpm. With one of the three tanks on standby, the system is capable of delivering 1400 gpm to satisfy primary system requirements. The standby service tank was available for service whenever the effluent conductivity of the inservice tanks rose to an unacceptable level. Each of the three demineralizer tanks normally contained 45-50 ft^3 of pre-regenerated mixed resins with a cation/anion ratio of 2 to 1. The three service tanks are designed for 400 psig operation, and normal flow is supplied by the condensate pumps. A circulating pump is provided to circulate water through the standby demineralizer tank prior to placing it into service.

The regeneration portion of the system consists of a cation regeneration and resin separation tank, an anion regeneration and resin mix tank, a sluice water tank, a sluice water pump, and acid tank and pump, a caustic tank and pump, and interconnecting piping.

System Status

The regeneration portion of this system has not been in use for many years, as pre-regenerated resins are used in this system; however, the equipment remains in place.

The system is presently maintained operational with resins in at least one service tank for flushing. Eventually, the system will be placed in layup.

5.2.26 Steam Turbine

The turbine is a high pressure, condensing, reaction, tandem compound, reheat 3600 rpm unit rated at 60,000 KW with the following steam conditions: 1250 psig, 547°F, exhausting at 1.0" Hg Absolute. The turbine consists of a high pressure and intermediate pressure and a low pressure element.

System Status

The turbine is periodically placed on the turning gear to maintain the shaft straight.

5. PLANT STATUS - (cont'd)

5.2.27 60-Megawatt Generator

The 60-Mw generator is a high-speed turbine-driven wound-rotor machine that is rated at 76,800 kva, 85 percent P.F., 3600 rpm, 60 cycle, 3 phase, 13,800v A-C, and 3213 amp. The generator is cooled by a hydrogen system, lubricated by a forced-flow lubricating system, and excited by a separate exciter attached to the end of the generator shaft through a reduction gear. A reserve exciter is provided.

System Status

The generator casing has been filled with nitrogen and the exciter brushes have been removed.

5.2.28 Turbine Oil and Hydrogen Seal Oil System

The Turbine Bearing Oil System receives cooled oil from the lube oil coolers to supply the necessary lubricating and cooling oil (via a bearing oil pressure regulator) to the turbine and generator bearings, exciter bearings, and exciter reduction gear. During normal operation, the necessary oil pressures are provided by the attached lube oil pump. During startup and shutdown, an ac motor-driven auxiliary lube oil pump provides oil pressure. Backup protection consists of the ac turbine bearing oil pump and the dc emergency bearing oil pump.

The Hydrogen Seal Oil System receives cooled oil from the lube oil coolers; and it supplies this oil, via a pressure regulator, to the inboard and outboard hydrogen seals of the generator. Backup protection is provided in the event the normal supply pressure drops or is lost, with an ac hydrogen seal oil pump and a dc emergency hydrogen seal oil pump.

Flexibility of the Turbine Oil Transfer System is brought about by the piping arrangement that allows the lubricating oil to be transferred or purified from several sources. With the lube oil transfer pump, turbine oil may be transferred from the lube oil reservoir to either the clean oil or dirty oil tanks located in the oil storage room.

System Status

This system is presently maintained operational to provide for the layup requirements of the turbine-generator.

5.2.29 Heating, Ventilation, and Air-Conditioning Systems

The Reactor Building ventilation system utilizes two 30-ton, 12,000-cfm air conditioning units for drawing fresh air into the building and for circulating the air throughout the building. The air enters the Reactor Building through two 20-inch isolation dampers in series, and is exhausted from the building by a centrifugal exhaust fan which has a capacity of 6000 cfm at 4 inches of water static pressure. The exhaust fan discharges through two 20-inch isolation dampers in series to the tunnel.

A 20-inch damper is also provided for recirculation of the exhaust fan discharge air. The exhaust system is provided with conventional and high-efficiency filters and with a gaseous and particulate radiation monitor system.

The Waste Treatment Building ventilation is provided by a 2000-cfm exhaust fan that draws air from the shielded vault areas of the building and exhausts the air through a duct out the floor of the building to the waste gas storage vault. The stack blowers then exhaust the air from the waste gas storage vault through the connecting tunnel and discharge the air up the stack.

The exhaust air from the Reactor Building and from the Waste Treatment Building are discharged into the tunnel connecting the Waste Treatment Building, the Reactor Building, and the Turbine Building to a plenum at the base of the stack. The stack is 350 feet high and is of structural concrete with an aluminum nozzle at the top. The nozzle tapers to 4 feet 6 inches at the discharge, providing a stack exit velocity of approximately 70 fps with the two 35,000 cfm stack blowers in operation.

The Turbine Building heating system provides heat to the turbine and machine shop areas through unit heaters and through automatic steam heating units.

The Control Room Heating and Air-Conditioning unit serves the Control Room, Electrical Equipment Room, Shift Supervisor's area, and Central Alarm Station.

The office area and laboratory are provided with a separate multi-zone heating and air-conditioning unit.

The heating boiler is a Cleaver-Brooks, Type 100 Model CB-189, 150-hp unit. At 150 psig, the boiler will deliver 6,275,000 Btu/hr. The boiler fuel is No. 2 fuel oil. The oil is supplied by and atomized in a Type CB-1 burner which will deliver 45 gph.

System Status

These systems are maintained operational and used as conditions require.

5.2.30 Waste Collection Systems

The functions of the Waste Collection and Treatment System are:

(1) To collect and store radioactive liquid waste generated in the plant.

(2) To collect and transfer depleted ion exchange resins to a shipping container.

(3) To process the collected waste as required for safe and economical disposal.

The Turbine Building Waste Collection System collects the liquid waste from the Turbine Building, the Waste Treatment Building, the waste gas storage vault, and the tunnel area in two storage tanks (one 4500 gal. and one 3000 gal.) located in the tunnel between the Reactor Building and the Turbine Building.

The Reactor Building Liquid Waste System consists of two retention tanks, each with a capacity of 6000 gallons, a liquid waste transfer pump, two sump pumps, and the necessary piping to route the waste liquid to the retention tanks and from the retention tanks out of the Reactor Building.

After a tank's contents are recirculated, a sample is withdrawn from the tank and analyzed for radioactivity concentrations prior to discharge.

The total amount of liquid waste discharged to the circulating water discharge line is measured with a flow totalizer water meter which can handle flow rates up to 100 gpm and a flow rate monitor with Control Room readout.

Depleted ion exchange resins are sluiced into a spent resin receiving tank in the Waste Treatment Building. The resins are then transferred to the spent resin liner. When the liner is full, it is dewatered and shipped offsite for disposal.

System Status

The Waste Collection Systems are maintained operational.

5.2.31 Fuel Transfer Bridge

The fuel transfer bridge is a specially-designed structure which is power-driven north and south on rails recessed in the floor at elevation 701'0" of the Containment Building.

The bridge traverses over the areas where service operations are performed at the reactor cavity, transfer canal, spent fuel storage pool and the new fuel storage racks. The bridge serves as the structural support for the fuel transfer hoist, and it provides an operating platform for personnel.

System Status

The fuel transfer bridge is normally positioned over the Fuel Element Storage Well and will be tested prior to use.

5.2.32 Communications Systems

The communications systems installed or otherwise available in the plant are:

(1) Central office trunk line telephone service for off-plant local and long distance calls.

(2) PABX (Private Automatic Branch Exchange) for interplant and intraplant calls and for off-plant calls to or from the site.

(3) Paging system for in-plant and site calls.

(4) Sound-powered telephone circuits, for in-plant voice communications.

(5) DPC ultra high-frequency radio network, for voice communications within DPC systems and headquarters, including mobile units.

(6) Microwave system for calls between LACBWR, Genoa No. 3, La Crosse, Alma, and to local numbers in La Crosse.

(7) ENS (Emergency Notification System) network for operational information transmission to NRC during incidents.

(8) Health Physics Network (HPN) for health physics information transmission to NRC during an emergency.

(9) NAWAS (National Warning System).

(10) Portable transceivers (handie-talkie) for mobile interplant and site voice communication.

System Status

The various communications systems are presently maintained operational. NAWAS will be removed from service after approval of the revised Emergency Plan.

5. PLANT STATUS - (cont'd)

5.2.33 Electrical Power Distribution

5.2.33.1 Normal ac Distribution

Oil Circuit Breaker 152R1 (25NB4) supplies the reserve auxiliary transformer located in the LACBWR switchyard.

Air Circuit Breakers 252R1A and 252R1B supply the 2.4-kv Bus 1A and Bus 1B from the 69/2.4-KV reserve transformer.

The 2400/480-volt Auxiliary Transformers 1A and 1B receive their power from the 2400-volt Buses 1A and 1B through Air Circuit Breaker 252AT1A from Bus 1A to Transformer 1A, and through Air Circuit Breaker 252AT1B from Bus 1B to Transformer 1B. The auxiliary transformers supply the 480-volt Buses 1A and 1B through Air Circuit Breaker 452M1A for Bus 1A and through Air Circuit Breaker 452M1B for Bus 1B.

The 480-volt buses supply larger equipment directly. They also supply motor control centers which furnish power to motors and other associated equipment connected to them through their respective breakers, including Motor Control Center (MCC) 120-volt ac Distribution Panels which supply 120-volt ac to equipment and instrumentation, excluding that required from a non-interruptible source.

The regular lighting cabinets are supplied from 480-volt buses 1A and 1B.

5.2.33.2 480-V Essential Buses 1A and 1B

The 480-v Essential Bus 1A Switchgear is normally supplied with electrical power from the 480-v Bus 1A through Breaker 452-52A. In the event of a loss of station power, the 480-v Essential Bus 1A is supplied with electrical power from Emergency Diesel Generator 1A through Breaker 452 EGA. Breakers 452-52A and 452 EGA are electrically interlocked to prevent both sources from supplying the bus.

The 480-v Essential Bus 1B Switchgear is normally supplied with electrical power from the 480-v Bus 1B through Breaker 452-52B. In the event of a loss of station power, the 480-v Essential Bus 1B is supplied with electrical power from Diesel Generator 1B through Breaker 452 EGB. Breakers 452-52B and 452 EGB are electrically interlocked to prevent both from supplying the bus.

The 480-v Essential Buses 1A and 1B may be cross-connected through the 480-v Essential Bus Tie Breakers 452 TBA and 452 TBB.

5.2.33.3 Emergency Diesel Generators 1A and 1B

The 1A Diesel Generator set system consists of a 250-kw diesel generator, a day tank fuel supply, a fuel transfer pump, a diesel-oil storage tank, a remote radiator and fan, a 100-kw test load, a local engine instrument panel,

a local generator panel, and a remote selector switch and alarms in the Control Room. The Diesel Generator set is located in the emergency generator cubicle which is on the grade floor level adjacent to the Machine Shop.

The function of the 1A Diesel Generator is to supply emergency power to the 480-v Essential Bus 1A which, in turn, supplies power to the Turbine Building MCC 1A, the Turbine Building 120-v Bus, the Turbine Building 120-v Regulated Bus, the Reserve Feed to the 120-v Non-Interruptible Bus, and the Reactor Plant Battery Charger.

The 1B Diesel Generator System consists of a 400 kw diesel driven generator, a 300-gallon fuel oil day tank, a 5500-gallon fuel oil storage tank, fuel oil transfer system and external remote radiator and fan, a 300 kw fan-cooled test load, a local engine control and instrument cabinet, and remote instrumentation and controls in the Control Room. The diesel generator set is located in the Generator Room of the Diesel Building which is south of the Electrical Penetration Room at elevation 641 feet.

The function of the 1B Diesel Generator is to supply emergency power to the 480-v 1B Essential Bus, which in turn supplies power to the Reactor MCC 1A 480-v Bus, Diesel Building MCC 480-v Bus, and the vital loads supplied by these MCC.

5.2.33.4 <u>120-V Non-Interruptible Buses</u>

The 120-v Non-Interruptible Buses maintain a continuous non-interruptible power supply to a portion of the essential reactor instrumentation and reactor control system circuits.

The 120-v Inverter 1A is designed for 3 KVA output and is powered by 125-v dc from the Reactor Plant Battery Bank through the Reactor Plant dc Distribution Panel. An automatic transfer switch is provided which will transfer the output to an alternate 120-v ac source in the event the inverter or its dc source fails. The alternate source for Inverter 1A is the Turbine Building 120-v Regulated Bus. The Inverter 1A is located in the Electrical Equipment Room.

The 120-v Non-Interruptible Bus 1B can be supplied with power from three sources. The normal main feed power source is supplied by Static Inverter 1B. The 5 KVA 1B Static Inverter is powered by 125-v dc from the Diesel Building Battery Bank through the Diesel Building 125-v dc Distribution Panel. Its alternate source is the Diesel Building MCC 480-v Bus through a static switch. The reserve feed power source is supplied by the Turbine Building 120-v Regulated Bus, through a breaker on TB MCC 1A, that is used only when the Static Inverter 1B is out of service.

The 120-v Inverter 1C is powered by 125-v dc from the Generator Battery Bank through the Generator Plant dc Distribution Auxiliary Panel. An alternate 120-v ac source is supplied through a breaker on Turbine Building MCC 1A through a static switch in the inverter.

5.2.33.5 125-V dc Distribution

The 125-v dc Distribution Systems supply dc power to all Generator Plant, Reactor Plant, and Diesel Building equipment requiring it.

The 125-v dc Distribution Systems are divided into three separate and independent systems each with its own battery, battery charger, and distribution buses. The buses may be cross-connected but are normally isolated from each other.

For each system, the battery charger provides the normal dc supply with the battery as the reserve supply. The battery floats on the line maintaining a full charge, and provides emergency dc power in the event of a loss of ac power to the battery charger or failure of the charger.

System Status

The Electrical Power Distribution System is maintained operational and required surveillance tests are performed on the Emergency Diesel Generators and 125-v batteries.

Systems will be evaluated in the future to combine or reduce redundancy of various loads, thereby reducing the number of buses, batteries, battery chargers, inverters, etc.

5.2.34 Post-Accident Sampling Systems

The Post-Accident Sampling Systems (PASS) are designed to permit the removal for analysis of small samples of either Containment Building atmosphere, reactor coolant, or stack gas when normal sample points are inaccessible following an accident. These samples will aid in determining the amount of fuel degradation and the amount of hydrogen buildup in containment. Samples will be removed to the laboratory for analysis.

5.2.34.1 Containment Atmosphere PASS System Description

The Containment Atmosphere Post-Accident Sampling System consists of a vacuum pump which takes a suction on the containment atmosphere at the 714' level. The atmosphere sample is drawn through two solenoid operated isolation valves, a heat exchanger, and moisture trap. Then the sample is discharged to the two in-parallel hydrogen analyzers with preset flowmeters; then either through a bypass line or a remote sample cylinder and back to the containment atmosphere at the 676' level through two solenoid operated isolation valves.

5.2.34.2 Stack Gas PASS System Description

The Stack Gas Post-Accident Sampling System makes use of the same equipment that provides the normal stack gas sample flow. The vacuum pump for stack gas sampling draws the extra flow, above what the SPING-4 and SPING-3 stack monitors draw, to make the total flow isokinetic to the stack discharge. This flow can be diverted through the post-accident sample canister by opening manual isolation valves. The sample canister is connected to the system by two quick disconnects, and, therefore, can be easily removed from the system and taken to the laboratory for analysis. The sample canister diversion valve is controlled from the local control panel in the No. 3 Feedwater Heater area.

5.2.34.3 Reactor Coolant PASS System Description

The Reactor Coolant Post-Accident Sampling System takes primary coolant from an incore flux monitoring flushing connection, through 2 solenoid-operated isolation valves with a heat exchanger between them, to a motor-operated pressure reducing valve. Downstream of the pressure reducing valve, the coolant sample can be diluted with demineralized water which then flows through the sample cylinder or its bypass valve, through another solenoid isolation valve, and back to the Containment Building basement or to the waste water tanks.

System Status

The Stack Gas and Containment Atmosphere PASS Systems will be maintained operational if the need exists. The Reactor Coolant PASS System is no longer needed.

5. PLANT STATUS - (cont'd)

5.3 RADIONUCLIDE INVENTORY ESTIMATES

Testing was conducted inhouse, using Health Physics personnel, to determine the location and the quantities of the radionuclides present at LACBWR. Several different types of samples and sampling techniques were used to qualify/quantify the radionuclide inventory. Each method will be described in its separate section. All samples were gamma scanned using HPGe detectors coupled to an ND6600 gamma spectroscopy computer system. This equipment has been calibrated to NBS traceable sources and is checked periodically to maintain this calibration.

5.3.1 Soil Sampling

A random location sampling of soil was conducted within the owner controlled area to determine if there was any ground contamination and, if so, to what extent it was present. Once a sample location was determined, a sample was taken from the surface down to a depth of 5-6 cm. Another sample was then taken in the same location to a depth of 10-12 cm.

The soil samples were placed in a 500 ml disposable marinelli for gamma analysis. These samples weighed between 500 and 800 grams. Each sample was gamma analyzed using the environmental HPGe and the ND6600. A three-hour count was used. This count was background corrected using a background soil sample taken approximately 15 miles north of the plant. Figures 5.1 and 5.2 show the sampling locations. Table 5-2 indicates the sampling results found. The sample depth can be determined by the sample identification. Example: Location 1 is surface to 5-6 cm; Location 2 is 5-6 cm to 10-12 cm.

TABLE 5-2

GROUND SAMPLE SURVEY RESULTS

Sample Location	Sample Activity in uCi/gram: Co-60	Cs-137	Mn-54	Cs-134	Co-58	Co-57
1-1	4.8E-7 ± 2.2E-8	2.6E-7 ± 1.8E-8	—	—	—	—
1-2	4.4E-7 ± 2.2E-8	2.5E-7 ± 2.1E-8	1.9E-8 ± 6.7E-9	—	—	—
2-1	1.4E-6 ± 6.2E-8	1.7E-6 ± 6.2E-8	—	—	—	—
2-2	1.2E-6 ± 3.1E-8	3.2E-6 ± 4.1E-8	—	6.0E-8 ± 1.7E-8	—	—
3-1	—	—	—	—	—	—
3-2	—	—	—	—	—	—
4-1	3.5E-8 ± 9.4E-9	3.5E-8 ± 1.1E-8	—	—	—	—
4-2	1.6E-8 ± 1.0E-8	—	—	—	—	—
5-1	3.0E-8 ± 1.3E-8	5.2E-8 ± 1.5E-8	—	—	—	—
5-2	—	4.5E-8 ± 1.4E-8	1.6E-8 ± 1.1E-8	—	—	—
6-1	7.6E-7 ± 3.0E-8	7.5E-7 ± 2.9E-8	3.0E-8 ± 9.0E-9	—	—	—
6-2	5.7E-8 ± 1.6E-8	7.4E-7 ± 1.7E-8	—	—	2.0E-8 ± 5.7E-9	—
7-1	5.6E-8 ± 1.9E-8	2.1E-8 ± 1.5E-8	—	—	—	—
7-2	—	—	—	—	—	—
8-1	7.3E-6 ± 8.0E-8	1.4E-5 ± 9.1E-8	—	1.5E-7 ± 2.8E-8	—	—
8-2	3.4E-6 ± 5.5E-8	1.1E-5 ± 7.6E-8	2.5E-8 ± 1.1E-8	9.7E-8 ± 2.1E-8	—	—
9-1	7.9E-7 ± 2.6E-8	5.7E-7 ± 2.4E-8	—	—	—	—
9-2	5.9E-8 ± 1.6E-8	3.7E-7 ± 2.2E-8	—	—	—	—
10-1	3.2E-7 ± 2.0E-8	1.5E-7 ± 1.2E-8	—	—	—	—
10-2	5.3E-8 ± 1.6E-8	1.1E-7 ± 1.2E-8	2.1E-8 ± 6.8E-9	—	—	—
11-1	1.8E-7 ± 1.1E-8	3.4E-7 ± 1.6E-8	—	—	—	—
11-2	7.6E-8 ± 8.6E-9	4.8E-7 ± 1.7E-8	1.2E-8 ± 4.3E-9	—	—	—
12-1	2.3E-7 ± 2.0E-8	2.9E-7 ± 2.4E-8	—	—	—	1.3E-8 ± 4.6E-9
12-2	—	1.3E-7 ± 1.6E-8	—	—	—	—

TABLE 5-2 - (cont'd)

Sample Location	Sample Activity in uCi/gram					
	Co-60	Cs-137	Mn-54	Cs-134	Co-58	Co-57
13-1	2.4E-7 ± 1.9E-8	6.5E-8 ± 2.2E-8	2.5E-8 ± 7.8E-9	---	---	---
13-2	4.9E-8 ± 1.6E-8	4.7E-8 ± 1.8E-8	3.2E-8 ± 8.6E-9	---	---	---
14-1	---	4.2E-8 ± 1.3E-8	---	---	---	---
14-2	---	9.6E-8 ± 1.3E-8	---	---	---	---
15-1	---	7.0E-7 ± 2.4E-8	---	---	---	---
15-2	---	5.9E-7 ± 2.3E-8	---	---	---	---
16-1	---	7.1E-7 ± 2.4E-8	---	---	---	---
16-2	---	3.8E-8 ± 1.7E-8	---	---	---	---
17-1	---	3.8E-7 ± 2.0E-8	---	---	---	---
17-2	---	2.6E-7 ± 1.7E-8	---	---	---	---
18-1	---	2.0E-8 ± 1.5E-8	---	---	---	---
18-2	---	2.5E-8 ± 1.5E-8	---	---	---	---
19-1	2.4E-7 ± 2.2E-8	1.1E-7 ± 1.6E-8	---	---	---	---
19-2	6.7E-8 ± 1.4E-8	---	---	---	---	---
20-1	---	4.7E-8 ± 1.2E-8	---	---	---	---
20-2	---	---	---	---	---	---
21-1	---	4.7E-8 ± 1.1E-8	---	---	---	---
21-2	---	1.0E-8 ± 9.6E-9	---	---	---	---
22-1	---	---	---	---	---	---
22-2	---	2.2E-8 ± 1.4E-8	---	---	---	---
Soil Background	2.0E-8 ± 4.1E-9	2.7E-8 ± 3.0E-9	4.9E-9 ± 2.2E-9	---	---	---

5. PLANT STATUS - (cont'd)

TABLE 5-2 - (cont'd)

Sample Location	Sample Activity in uCi/gram					
	Co-60	Cs-137	Mn-54	Cs-134	Co-58	Co-57

Several points to the north of LACBWR were resampled to verify activity results. The following is a list of these resamples.

10-1 (A)	1.9E-7 ± 2.3E-8	5.8E-8 ± 1.8E-8	--	--	--	--
11-1 (A)	4.2E-7 ± 2.3E-8	7.7E-7 ± 2.1E-8	--	--	--	--
12-1 (A)	1.7E-7 ± 2.2E-8	3.2E-7 ± 2.1E-8	--	--	--	--
13-1 (A)	--	5.1E-8 ± 1.4E-8	--	--	--	--

5.3.2 Plant Contamination

A plant smear survey was performed of all accessible interior building surfaces in an attempt to determine the amount of loose surface contamination in the plant. The specific isotopic identification of the contamination was also determined. Each smear was gamma scanned to determine not only a correlation factor in uCi per DPM/100 cm^2 but also the percentile of each radioisotope present in the mixture. From previous analysis, it has been determined that Fe-55 is the major beta emitter in the plant and is in approximately the same percentage present as Co-60. Fe-55 will be the only beta emitting isotope listed as a contaminant. Alpha activity on the surfaces has been checked by the use of an Internal Proportional Counter and has been found to be negligible and so will not be considered. The survey did indicate that the major isotopes present in the plant's loose surface contamination are the following isotopes:

Isotope	1/2 life
Co-60	5.27 years
Cs-137	30.1 years
Mn-54	312.2 days
Fe-55	2.7 years

It must be realized when reviewing the results of this survey that a 100% smearing of plant surfaces was not performed and therefore the following data is subject to significant error. Table 5-3 lists the isotopes found in the plant's loose surface contamination and the total uCi content of loose surface contamination on interior building surfaces.

TABLE 5-3

PLANT LOOSE SURFACE CONTAMINATION

Location	Isotopes Present, in uCi							Total Area uCi Content
	Cs-60	Cs-137	Mn-54	Ce-144	Co-57	Cs-134	Fe-55	
Turbine Building (TB)								
a) Main Floor	0.83	0.07	—	—	—	—	0.83	1.73
b) Mezzanine - including stop valve area	0.49	0.14	0.04	—	—	—	0.49	1.16
c) Grade Floor - includes feedwater heater area	0.42	0.06	0.02				0.42	0.92
d) Tunnel	0.81	0.18	0.06	—	—	—	0.81	1.86
Containment Building (CB)								
a) Above grade	3.16	0.2	0.39	—	—	—	3.16	6.91
b) Below grade	31.44	7.40	2.36	0.04	0.04	0.08	31.44	72.80
Waste Treatment Building	7.57	0.48	0.66	—	—	—	7.57	16.28
Totals	44.72	8.53	3.53	0.04	0.04	0.08	44.72	101.66

5.3.3 Plant System Contamination Radionuclide Content

The internal surfaces of many inplant systems were exposed to radionuclide contaminants during plant operation. It was determined to be desirable to identify the amount and type of system contamination for the initial entry into SAFSTOR. This data can then be used to assist in monitoring LACBWR's radioisotopic content during its SAFSTOR period. To obtain the most accurate analysis of these systems, piping/component destruction would be necessary. This was determined to be an undesirable method of analysis at this time.

A method of nondestructive sampling was developed. Each system that would be sampled was looked at and a nondestructive entry point was found. Once the system was opened, the piping was dried and a 1 cm^2 area was scraped to bare metal. As the scraping was being done, the corrosion layer that was being removed was vacuumed into a glass fiber filter 47 mm in diameter. This filter was then gamma scanned for radionuclide identification using the Nuclear Data computer-based gamma spectroscopy unit connected to a GeLi detector.

A sample of the crud layer from the bottom of the FESW was obtained using a vacuum system. This crud was mixed with the water and an aliquot was taken and gamma scanned. This was used to determine the activity in the FESW crud layer. Because of the inaccessibility of the bottom of the reactor vessel, the gamma spectrum obtained from the FESW sample was used as the gamma spectrum for any crud layer in the vessel bottom. Conversation with representatives from other facilities who have looked at their reactor vessel bottom indicates a very small crud layer in the vessel.

Each sample was also alpha counted to determine the total alpha content. The individual alpha isotopic mixture was not determined. Alpha analysis was performed with a Canberra Internal Proportional counter.

In systems where the piping/component radiation levels varied significantly, the radiation level of the sample area was found. The remainder of the system was surveyed and the piping/component area was classified by radiation level. This technique allowed the sampled area uCi/cm^2 value to correspond with a radiation level. As the system radiation level varied, a uCi/cm^2 radiation level value was used to proportion the system areas to that of the sample area, thus allowing a total system uCi content to be determined. After sampling, each system was returned to an as-found condition.

Fe-55, as stated in the previous section, will be the only beta emitting isotope to be considered in the plant isotopic content. This will again be in the same percentage as that of the Co-60 found in the systems. As in the plant contamination survey performed, the same isotopic composition occurs in the piping systems with the major nuclides listed as follows:

Isotope	1/2 life
Co-60	5.27 years
Cs-137	30.1 years
Mn-54	312.2 days
Fe-55	2.7 years

As with the plant contamination sampling program, this testing of piping systems is not an absolute value. Significant error can be associated with this sampling technique due to the inability to actually analyze a system component or pipe. Table 5-4 lists the isotopes, uCi of each isotope and total uCi content found throughout the LACBWR piping systems.

Table 5-5 is a listing of the radionuclide quantities determined to be located throughout LACBWR systems. The values listed are subject to error due to the required non-destructive sampling techniques used and the inability to sample all components and areas.

TABLE 5-4

PLANT SYSTEMS NUCLIDE CONTENT IDENTIFICATION

Plant System	Nuclide Activity, in uCi									System Total
	Fe-55	Alpha	Co-60	Cs-137	Mn-54	Cs-134	Nb-95	Co-57	Zn-65	uCi Content
CB Ventilation	1.6 E3	—	1.6 E3	1.7 E2	1.3 E2	1.4	—	—	—	3.5 E3
Offgas - upstream of filters	6.0 E2	—	6.0 E2	4.4 E4	1.0 E2	—	—	—	—	4.5 E4
Offgas - downstream of filters	6.5 E2	—	6.5 E2	8.3 E2	—	—	—	—	—	2.1 E3
TB drains	1.7 E4	4.0 E1	1.7 E4	5.0 E3	7.8 E2	—	—	—	—	4.0 E4
CB drains	3.8 E4	3.2	3.8 E4	2.4 E3	2.6 E3	—	—	—	—	8.1 E4
TB Waste Water	3.6 E3	6.8	3.6 E3	1.2 E2	1.5 E2	—	5.6 E1	—	—	7.5 E3
CB Waste Water	2.1 E5	7.9 E1	2.1 E5	2.3 E3	1.7 E4	—	—	1.3 E2	1.4 E3	4.4 E5
Main Steam	2.6 E5	2.9 E2	2.6 E5	—	2.0 E4	—	1.0 E3	—	—	5.4 E5
Turbine	9.3 E2	1.8	9.3 E2	2.0 E2	1.8 E2	—	—	—	—	2.2 E3
Primary Purification	8.9 E4	1.2 E1	8.9 E4	—	8.8 E3	—	—	—	2.1 E3	1.9 E5
Emergency Core Spray	1.8 E3	5.0	1.8 E3	1.1 E 2	1.4 E2	—	—	—	—	3.9 E3
Overhead Storage Tank	1.3 E4	3.4 E1	1.3 E4	7.8 E2	9.5 E2	—	—	—	—	2.8 E4
Seal Inject	1.6 E3	3.8	1.6 E3	5.5 E1	1.5 E2	—	—	7.5	—	3.4 E3

5. PLANT STATUS - (cont'd)

TABLE 5-4 - (cont'd)

Plant System	Nuclide Activity, in uCi								System Total
	Fe-55	Alpha	Co-60	Mn-54	Co-57	Co-58	Zn-65	Other	uCi Content
Decay Heat	1.0 E5	4.9 E2	1.0 E5	3.1 E4	1.6 E2	3.2 E3	3.5 E3		2.4 E5
Boron Inject	1.4 E5	6.6 E2	1.4 E5	4.2 E4	2.1 E2	4.3 E3	4.7 E3		3.3 E5
Reactor Coolant PASS	9.9 E3	4.6 E1	9.9 E3	2.9 E3	1.5 E1	3.0 E2	3.3 E2		2.3 E4
Alternate Core Spray	2.0 E4	9.4 E1	2.0 E4	5.9 E3	3.0 E1	6.1 E2	6.7 E2		4.7 E4
Shutdown Condenser	2.3 E5	1.1 E3	2.3 E5	6.9 E4	3.5 E2	7.1 E3	7.8 E3		5.5 E5
Control Rod Drive Effluent	1.5 E5	7.2 E2	1.5 E5	4.6 E4	2.3 E2	4.7 E3	5.1 E3		3.6 E5
Forced Circulation	1.5 E6	7.0 E3	1.5 E6	4.4 E5	2.3 E3	4.5 E4	5.0 E4		3.5 E6
Reactor Vessel and Internals	2.5 E6	1.2 E4	2.5 E6	7.6 E5	3.9 E3	7.8 E4	8.6 E4		5.9 E6
Condensate after beds & Feedwater	2.1 E5	2.8 E2	2.1 E5	3.2 E4	—	1.6 E3	3.1 E3		4.6 E5
Condensate to beds	3.9 E4	3.1 E1	3.9 E4	1.3 E4	1.5 E1	6.3 E2	6.7 E2	Fe-59 = 5.2 E2 Nb-95 = 1.1 E2 Ru-103 = 4.9 E1 Ce-144 = 1.2 E2	9.3 E4

TABLE 5-4 - (cont'd)

Plant System	Nuclide Activity, in uCi								System Total
	Fe-55	Alpha	Co-60	Mn-54	Cs-137	Ce-144	Zn-65	Other	uCi Content
Fuel Element Storage Well System	8.5 E5	3.9 E2	8.5 E5	1.4 E4	—	1.0 E4	—		1.7 E6
Fuel Element Storage Well - all but floor	1.3 E3	4.9	1.3 E3	6.0 E2	4.6 E3	—	4.5 E2		8.3 E3
Fuel Element Storage Well floor	2.6 E7	7.6 E3	2.6 E7	5.0 E5	4.1 E4	—	1.1 E5	Cs-134 = 1.3 E2 Co-58 = 1.3 E2	5.3 E7
Resin lines	1.3 E5	1.0 E2	1.3 E5	4.2 E4	—	4.0 E2	2.2 E3	Fe-59 = 1.7 E3 Co-57 = 4.8 E1 Co-58 = 2.1 E3 Nb-95 = 3.5 E2 Ru-103 = 1.6 E2	3.1 E5
Main Condenser	1.1 E7	8.5 E3	1.1 E7	3.6 E6	—	3.4 E4	1.9 E5	Fe-59 = 1.4 E5 Co-57 = 4.1 E3 Co-58 = 1.7 E5 Nb-95 = 3.0 E4 Ru-103 = 1.4 E4	2.6 E7

TABLE 5-5

TOTAL CONTAMINATION IN PLANT SYSTEMS

Isotope	uCi Content	% of Total	1/2 Life
Co-60	4.4 E7	46.4	5.27 years
Mn-54	5.7 E6	6.0	312.2 days
Co-57	1.2 E4	1.3 E-2	270 days
Co-58	3.2 E5	0.3	70.8 days
Zn-65	4.7 E5	0.5	243.8 days
Fe-59	1.4 E5	0.1	45.1 days
Nb-95	3.2 E4	3.4 E-2	35.2 days
Ru-103	1.4 E4	1.5 E-2	39.4 days
Ce-144	4.5 E4	4.7 E-2	284.2 days
Cs-137	1.0 E5	0.1	30.1 years
Cs-134	1.3 E2	1.4 E-4	2.1 years
Fe-55	4.4 E7	46.4	2.7 years
Alpha	5.4 E4	5.7 E-2	—
Total	9.5 E7 uCi		

5. PLANT STATUS - (cont'd)

5.3.4 Activated Metal Components

Reactor components in and near the reactor core during power operation become radioactive due to nuclear interaction with the large neutron flux present in this region. Most of the residual radioactivity is produced by n,γ reactions with the atomic nuclei of the target material although n,p reactions, for example the production of ^{14}C from ^{14}N, are also significant.

The residual radioactivity in the materials of the various LACBWR reactor components has been estimated using activation analysis theory and, where available, actual data from laboratory analyses of irradiated metal samples. Best estimates of neutron fluxes in and irradiation histories of specific components were used in these calculations. Original material chemical compositions were obtained from actual material certification records when readily available but standard compositions for specified materials were used in some cases. Radioactive decay of the activation product nuclides has been taken into account to obtain the best estimate values for the residual radioactivity as of January 1, 1988.

The results of these calculations are tabulated in Table 5-6. As of January 1, 1988, by far the largest contribution (210,000 Ci) to the radioactivity inventory in LACBWR activated metal components is the isotope ^{55}Fe (HL = 2.7y). The next largest contributor (101,000 Ci) is ^{60}Co (HL = 5.27y). The activity of these two relatively short-lived isotopes will decrease very significantly during the proposed SAFSTOR period. The major long-lived contributor to the radioactivity inventory (10,700 Ci) is ^{63}Ni (HL = 100y). The activity of other activation product nuclides have been lumped together in two categories, those with half lives less than 5 years and those with half lives greater than 5 years. The group with HL <5y consists mostly of ^{95}Zr (64d) in Zircaloy components and ^{51}Cr (27.7d) and ^{59}Fe (44.6d) in stainless steel components along with small quantities of ^{113}Sn (115d), ^{119}Sn (293d), ^{123}Sn (129d), ^{175}Hf (70d), ^{181}Hf (42.4d), ^{181}W (121d) and ^{185}W (75.1d). The group with HL >5y consists mostly of ^{59}Ni ($8x10^4$y) with small quantities of ^{14}C (5730y), ^{93}Zr ($1.5x10^6$y), ^{121}Sn (50y), ^{113}Cd (14.6y), ^{94}Nb ($2x10^4$y) and ^{99}Tc ($2.13x10^5$y).

TABLE 5-6

RADIOACTIVITY INVENTORY IN ACTIVATED METAL - JANUARY 1, 1988

	Estimated Curie Content					
				Other Nuclides		
Compon[illegible]	^{60}Co	^{55}Fe	^{63}Ni	$T_{1/2}$ <5y	$T_{1/2}$ >5y	Total
In Reac[illegible]						
Fuel [illegible] 8 SS)	22,109	63,221	1,352	2,810	15	89,507
Control R[illegible]	4,886	4,826	817	24	15	10,568
Core Vertical Posts (52)	1,270	594	63	2,396	4	4,327
Core Lateral Support Structure	9,108	21,477	770	105	8	31,468
Steam Separators (16)	33,439	78,851	2,826	386	30	115,532
Thermal Shield	1,443	3,402	123	17	1	4,986
Pressure Vessel	347	1,029	10	2	~0	1,388
Core Support Structure	6,458	15,230	546	75	6	22,315
Horizontal Grid Bars (7)	173	408	15	2	~0	598
Incore Monitor Guide Tubes	307	188	611	7	5	1,118
Total	79,540	189,226	7,133	5,824	84	281,807
In FESW						
Fuel Shrouds (24 SS)	13,667	14,988	2,384	~0	26	31,065
Fuel Shrouds (73 Zr)	918	1,007	95	27	3	2,050
Control Rods (10)	3,456	2,386	910	~0	17	6,769
Start-up Sources (2)	3,177	2,285	156	5	3	5,626
Total	21,218	20,666	3,545	32	49	45,510

5. PLANT STATUS - (cont'd)

5.4 RADIATION LEVELS

5.4.1 Plant Radiation Levels

Upon entering the initial phase of LACBWR's SAFSTOR mode, base line gamma radiation surveys were performed throughout the plant. General area radiation levels are listed below. These levels will be routinely monitored and tracked. Specific area hot spots will also be looked for and recorded on each area survey.

Area	General Area Gamma Radiation Levels
Containment Building:	
Shutdown Condenser Platform	10-20 mRem/hr
701' Level	6-12 mRem/hr
Mezzanine Level East	5-10 mRem/hr
Mezzanine Level West	20-30 mRem/hr
West Nuclear Instrument Platform	40-90 mRem/hr
East Nuclear Instrument Platform	10-20 mRem/hr
Purification Cooler Platform	5-10 mRem/hr
Grade Floor North and East	7-20 mRem/hr
Grade Floor West	75-120 mRem/hr
Upper Control Rod Drive Area	60-120 mRem/hr
Basement	10-40 mRem/hr
Primary Purification Demineralizer	7-17 mRem/hr
Retention Tank Area	250-400 mRem/hr
Lower Control Rod Drive Area	60-150 mRem/hr
Forced Circulation Pump Cubicles	150-400 mRem/hr
Turbine Building:	
Main Floor	<1-3 mRem/hr
Mezzanine	<1-4 mRem/hr
Stop Valve Area	10-85 mRem/hr
Grade Floor	1-10 mRem/hr
Feedwater Heater Area	5-20 mRem/hr
Tunnel	10-50 mRem/hr
Machine Shop	<1 mRem/hr
1B Diesel Room	<1 mRem/hr
Electrical Penetration Room	2-7 mRem/hr
Waste Treatment Building:	
Main Floor	1-20 mRem/hr
Basement	10-100 mRem/hr
Building Exteriors:	
Exterior of Waste Treatment Building	<1 except for south side where there is one spot between 3-4 mRem/hr
Exterior of Containment Building	<1 except for one spot on south side reading 7 mRem/hr

5.4.2 System Radiation Levels

During SAFSTOR the major radioactively contaminated systems at LACBWR will be monitored in order to trend system cleanups and radioactivity decay. A program consisting of 100 survey points located throughout the plant has been established. Initial system contact readings have been taken and will be monitored on a frequency determined to adequately trend any radiation level changes. The individual survey locations may change during the SAFSTOR period as plant parameters change.

The following is a list of the initial survey points and the associated dose rates.

Survey Point #	Survey Point Location	Contact Dose Rate (mrem/hr)
1	Condensate Line to and from OHST	25
2	Condensate Line to and from OHST	24
3	Condensate Line to and from OHST	33
4	1A Condensate Pump Discharge Line	12
5	Emergency Overflow Line	27
6	Emergency Overflow Bypass Line	33
7	Ice Melt Line	3
8	1A Reactor Feed Pump	16
9	Near 1B Reactor Feed Pump Discharge Valve	11
10	Side of #3 Feedwater Heater	26
11	Reheater Level Control Chamber	26
12	South End of Reheater	13
13	Gland Exhaust Condenser Loop Seal	35
14	Main Steam Line	48
15	Main Steam Line	50
16	Offgas System Flame Arrestor	8
17	1B Waste Water Pump	26
18	1A Waste Water Pump	60
19	End of 3000 Gallon Waste Tank	170
20	End of 4500 Gallon Waste Tank	120
21	Side of Gland Seal Steam Generator	1100
22	Side of Gland Seal Steam Generator	160
23	Main Steam Bypass Line	17
24	Turbine Inlet Valve Body	23
25	Main Steam Line	24
26	Reheat to Flash Tank Line	11
27	Flash Tank	5
28	Seal Injection Heater	31
29	#2 Feedwater Heater Bypass Line	100
30	Feedwater Heater Bypass Line	24
31	Bottom of Gland Exhaust Condenser	170
32	Top of Gland Exhaust Condenser	20
33	Condensate into Air Ejector Line	7

5. PLANT STATUS - (cont'd)

Survey Point #	Survey Point Location	Contact Dose Rate (mrem/hr)
34	Air Ejector	8
35	Low Pressure Turbine Manhole Cover	6
36	End of High Pressure Turbine	2
37	Primary Purification 1A Filter Inlet Line	38
38	Primary Purification Pump	140
39	Exhaust Ventilation Duct	9
40	Containment Bldg. Grade Level N Shield Wall	6
41	1A Fuel Element Storage Well Pump	70
42	1B Fuel Element Storage Well Pump	80
43	FESW Filter Discharge Line	180
44	FESW System Cooler	1000
45	Hydraulic Valve Actuation System Header	60
46	Base of Hydraulic Valve Accumulator	24
47	Wall at Electrical Penetration	30
48	Handrail on NW Nuclear Instrumentation (NI) Platform	100
49	Shield Wall on N NI Platform	4
50	Primary Purification to OHST Line	6
51	Above Primary Purification Cooler Inlet Valve	25
52	Cold Leg of Reactor High Level Transmitter Line	46
53	Seal Injection Resevoir	30
54	Reactor Cavity Drain Line	44
55	1A Core Spray Pump Discharge Line	10
56	Reactor Water Level Sightglass Line	180
57	Reactor Water Level Sightglass Line	100
58	Cont. Bldg. Mezzanine Level N Shield Wall	4
59	Steam Trap Cont. Bldg. Mezz. Level NW Wall	23
60	Fuel Element Storage Well Line	400
61	Fuel Element Storage Well Line	420
62	Fuel Element Storage Well Line	60
63	Fuel Element Storage Well Skimmer Line	90
64	Wall near Fuel Transfer Canal Drain	35
65	Relief Valve Platform at Level Transmitter Isolation	80
66	Shutdown Condenser	11
67	Shutdown Condenser Condensate Line	6
68	1B Retention Tank	300
69	1A Retention Tank	130
70	By Primary Purification Cation Tank	24
71	Decay Heat Cooler	25
72	Decay Heat Cooler	18
73	Decay Heat Cooler Bypass Valve	70
74	Decay Heat Pump Suction Line	32
75	Handrail at Shutdown Condenser Condensate Valves	28
76	Seal Injection DP Transmitter	44
77	Top of Upper Control Rod Drive Mechanism	370
78	Top of Upper Control Rod Drive Mechanism	200
79	Wire mesh screen on N Upper Control Rod Platform	22

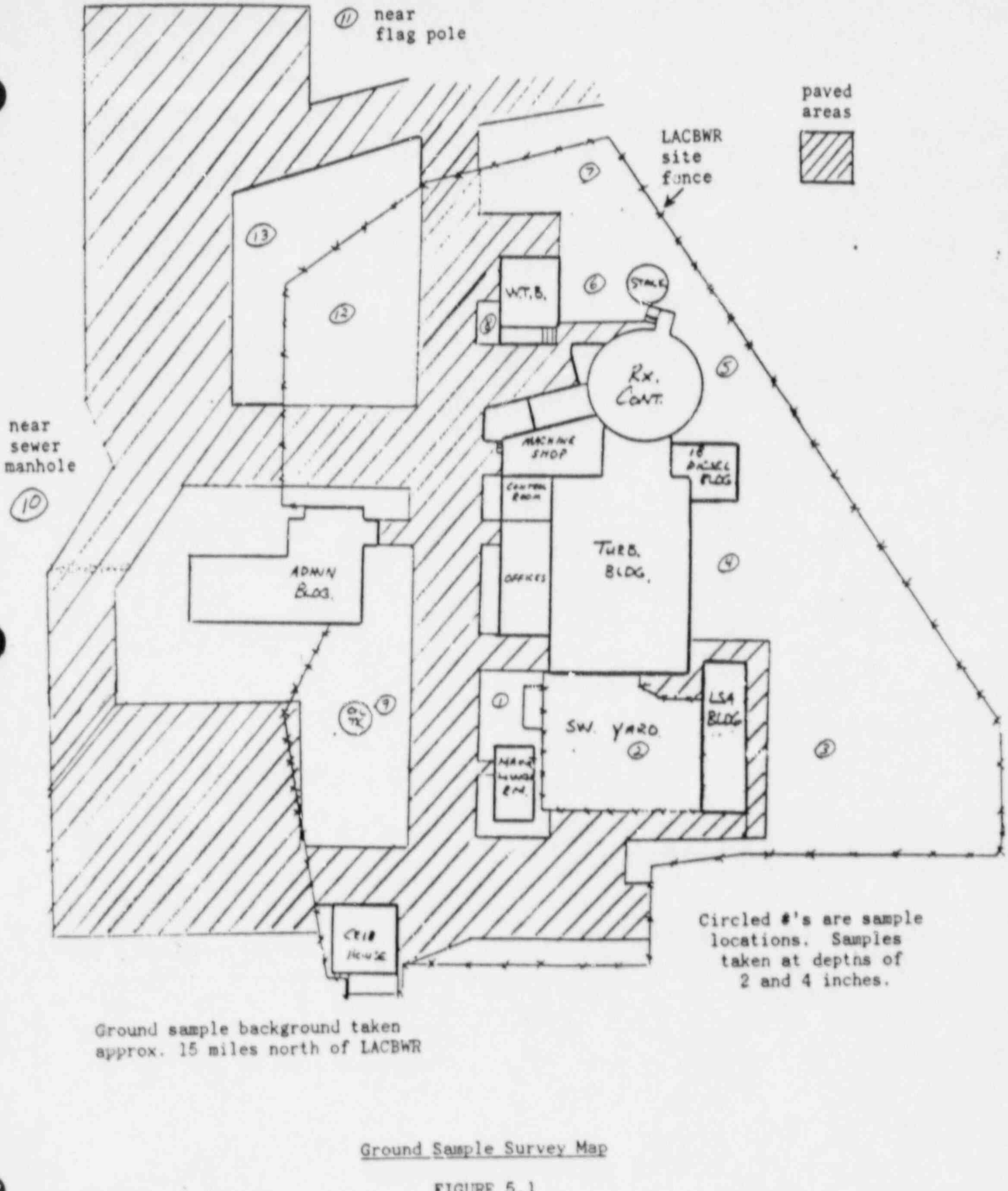
Survey Point #	Survey Point Location	Contact Dose Rate (mrem/hr)
80	Bottom of Upper Control Rod Drive Mechanism	1000
81	Top of Upper Control Rod Drive Mechanism	500
82	Bottom of Upper Control Rod Drive Mechanism	800
83	Effluent Lines on Upper Control Rod Platform	390
84	Sump Pump Discharge Line to Retention Tank	260
85	At Forced Circulation Pump Filters	33
86	Retention Tank Pump	60
87	Under Lower Control Rod Drive Mechanism	246
88	Control Rod Drive Hydraulic System Header	190
89	Decay Heat Pump	150
90	1B Forced Circulation Pump Suction Line	1000
91	1B Forced Circulation Pump Suction Line	1100
92	1A Forced Circulation Pump Suction Line	500
93	1A Forced Circulation Pump Suction Line	600
94	1A Forced Circulation Pump Discharge Line	700
95	Feedwater Line in Forced Circulation Cubicle	130
96	1A Forced Circulation Pump	130
97	Handrail at 1A Forced Circ. Pump Suction Line	250
98	1A Forced Circulation Pump Discharge Line	800
99	1A Forced Circulation Pump Discharge Line	600
100	1A Forced Circulation Pump Suction Line	700

5.5 PLANT PERSONNEL DOSE ESTIMATE

During normal/routine SAFSTOR operations at LACBWR, average whole body radiation dose received by plant personnel should be no more than 0.600 Rem per individual per quarter. This average dose is expected to decrease during the SAFSTOR period due to isotopic decay. Individual doses will be dependent upon work being performed. Plant personnel will not be allowed to exceed 2.7 Rem/quarter or 5.0 Rem/year.

5.6 SOURCES

As authorized by the facility license, sealed sources for radiation monitoring equipment calibration, reactor instrumentation, reactor startup, and fission detectors will continue to be possessed and/or used. Additionally, sources will be used as authorized without restriction to chemical or physical form for sample analysis, instrument calibration and/or as associated with radioactive apparatus and components.



Ground sample background taken approx. 15 miles north of LACBWR

<u>Ground Sample Survey Map</u>

FIGURE 5.1

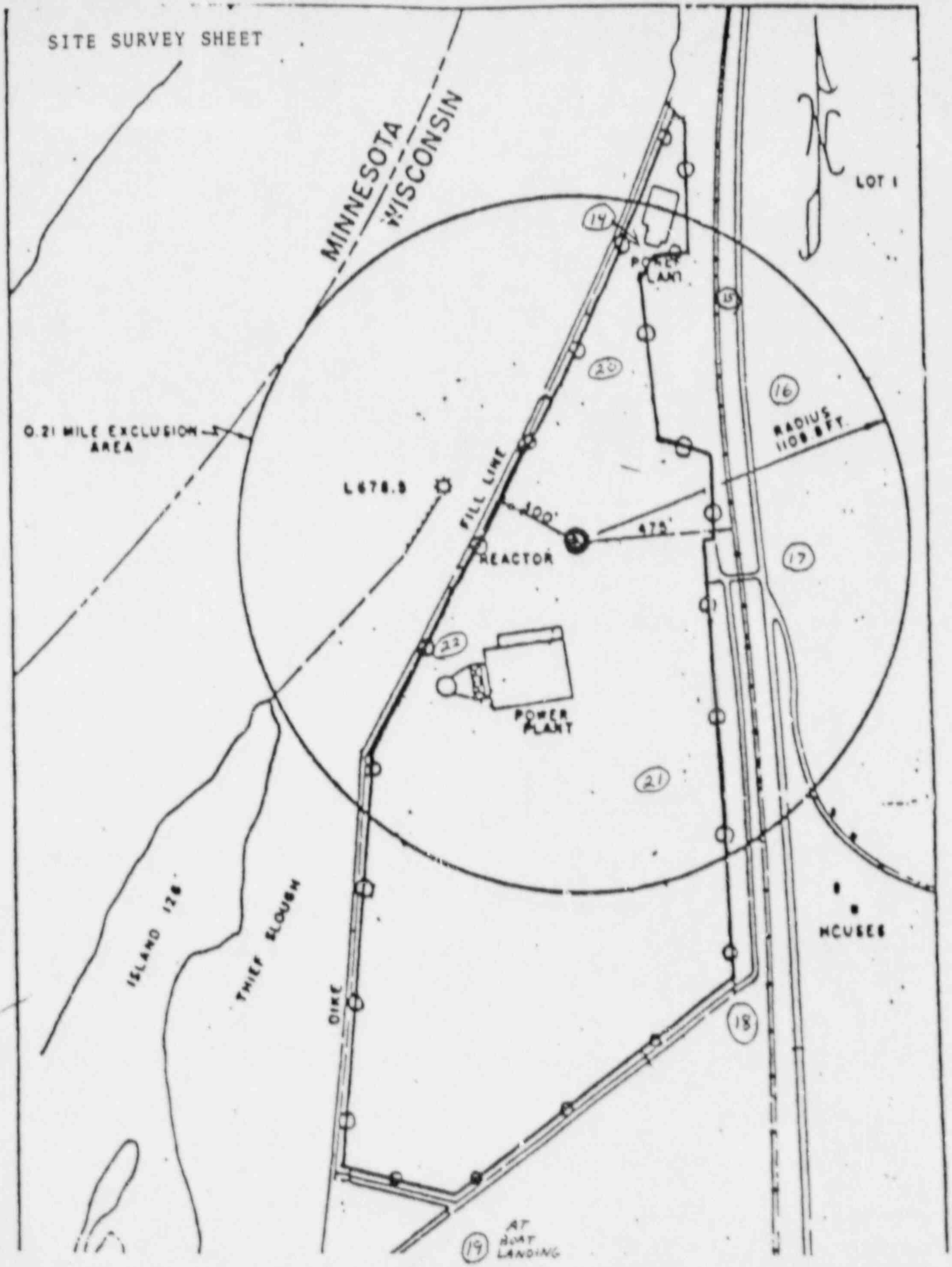

SITE SURVEY SHEET
MINNESOTA
WISCONSIN
LOT 1
14
POWER PLANT
15
20
16
RADIUS 1108.8 FT.
0.21 MILE EXCLUSION AREA
L676.5
FILL LINE
200'
475'
REACTOR
17
22
POWER PLANT
21
ISLAND 128
THIEF SLOUGH
HOUSES
DIKE
18
19
AT BOAT LANDING

FIGURE 5.2

6. DECOMMISSIONING PROGRAM

6.1 OBJECTIVES

The primary objective of the Decommissioning Program at LACBWR will be to safely monitor the facility and prevent any unplanned release of radioactivity to the environment. Some of the goals during the SAFSTOR period are as follows:

- To safely store activated fuel until it can be removed from the site.
- To establish a monitoring and surveillance program for comparison to baseline conditions.
- To maintain systems required during the SAFSTOR period.
- To lay up non-operating systems.
- To salvage equipment that is no longer being used.
- To handle radioactive waste generated during the SAFSTOR period in accordance with plant procedures and applicable requirements.
- To reduce general area radiation levels in the vicinity of equipment operated or maintained during the SAFSTOR period to limit personnel dose to as low as reasonably achievable.
- To start decontaminating and dismantling unused systems while minimizing the generation of radioactive waste and personnel dose from this activity.
- Maintain qualified and trained staff to fulfill these goals.

6.2 ORGANIZATION AND RESPONSIBILITIES

The organization of the SAFSTOR staff at LACBWR is as indicated in Figure 6-1. The staff may change as activities being performed vary and staffing needs change. The organization is directed by a Plant Superintendent, who reports directly to the Dairyland Power Cooperative Assistant General Manager for Operations. The four groups and three individuals which report directly to the Superintendent each have distinct functions in insuring the safety of the facility during the SAFSTOR mode.

The Superintendent is responsible for the safety of the facility, its daily operation and surveillance, long range planning, licensing and any other responsibilities which may come to light in long-term SAFSTOR operation. Quality assurance activities and security control and support are provided by a Cooperative-wide quality assurance and security program. The Plant Superintendent is responsible for operation of any onsite security required as well as insuring compliance with the quality assurance program.

The Operations and Instrument and Electrical Supervisor is responsible for the day-to-day activities of the operators, instrument technicians and electricians. This supervisor is responsible to insure that adequate staff is present to comply with the terms of the license, training commitments and responsibilities are met, and that the personnel reporting are fit for duty. The Operations and Instrument and Electrical Supervisor is responsible for coordination of all Technical Specifications required tests and problem reporting of any deficiencies to the Plant Superintendent.

The Shift Supervisor is responsible for operating the shift and insuring that the facility is maintained in a safe and efficient manner. The Shift Supervisor will direct and be responsible for all operations and maintenance activities occurring on shift. The Shift Supervisor will insure that routine rounds are made, logs are kept and equipment maintenance requests are properly initiated.

The Operators are responsible for the operation of the facility. They will ensure that all equipment is operated in a proper manner consistent with the license. When it is necessary to handle fuel, they will do so in compliance with their training certification and the procedures of the facility. The Operators will also be responsible to insure that procedural deficiencies they discover receive a prompt review by initiating necessary paperwork. The Operators will tour the facility and insure that the fuel storage well and its fuel, as well as all supporting systems, are in a clean, operable mode.

The Instrument Technicians will be responsible for maintaining the instrumentation within the facility necessary to safely store the expended fuel. They will perform all surveillance tests required as well as all maintenance requests initiated on instrumentation.

The Electricians will be responsible for maintaining all electrical equipment in operating systems in accordance with procedures and completing all maintenance requests and surveillance tests that are required. They will be responsible for any other equipment within the plant which may be used as backup or spares for operable systems or for backups for other facilities within the Dairyland system. They are also responsible for electrical breaker maintenance and such other responsibilities as may be assigned by supervision.

The Health and Safety Supervisor is responsible for the radiological health and safety of the general public in the area surrounding the plant as well as the safety of the staff and all visitors to the plant. The Health and Safety Supervisor will ensure that all long-term radiological and environmental surveillance programs in the SAFSTOR operation are carried out and that proper reports on radiation exposure throughout the facility are maintained. This individual will ensure that all radiation exposure controls are in place and ensure that contamination and daily, monthly and annual exposure limits on personnel are complied with. The Health and Safety Supervisor will be responsible for the ALARA program and will ensure that all personnel

stationed at or visiting LACBWR comply with it in spirit as well as regulation. This supervisor will also assign the day-to-day duties of the health physics technicians.

The Health Physics Technicians will be responsible for the radiation protection and chemistry programs at LACBWR. They will perform all tasks required for surveillance and will provide all work coverage required by special work permits. They will maintain as required the exposure records of personnel, take all the readings necessary to guard against the spread of contamination and provide input to the long-term radionuclide inventory program. They will report, as directed by the Health and Safety Supervisor, to the Duty Shift Supervisor as required.

The Mechanical Maintenance Supervisor is responsible for the assignment of mechanical maintenance duties and will direct the completion of all maintenance requests and surveillance tests of a mechanical nature. The Mechanical Maintenance Supervisor is responsible for the preventive maintenance program established on those systems necessary to maintain the SAFSTOR condition. This supervisor is responsible for overall maintenance on all of the plant equipment which may serve as backups to the required systems or backup supplies to the rest of the Dairyland system.

Maintenance Mechanics are responsible for the completion of all mechanical maintenance tasks. These tasks include all surveillance requirements and work requests defined in maintenance orders as well as general duties as assigned by the maintenance supervisor.

The Administrative Assistant is responsible for overall administration of LACBWR. The Administrative Assistant will maintain all records required under technical specifications from the period of operation and will maintain a record of all activities involved in facility shutdown and establishment of the SAFSTOR mode. This supervisor will direct the clerical staff and ensure that all clerical functions are performed adequately. The Administrative Assistant will maintain all budget expense and project accounts and will be the lead individual in preparing the LACBWR budget. Duties will also include assigning to staff personnel all required responses to regulatory agencies, other Dairyland departments, etc., and ensuring that these tasks are completed by the established deadline.

The Secretaries will be responsible for all clerical tasks at LACBWR, including word processing and the personal computer database. They will also be responsible for the telephone communication switchboard operation and other tasks as assigned by the Administrative Assistant.

The Licensing Engineer will be responsible for all facility licensing. This will include steps preparatory to eventual shipment of SAFSTOR fuel and proceeding into the DECON mode. The Licensing Engineer will be the principal liaison on behalf of the Superintendent for the contact with the Nuclear Regulatory Commission and other regulatory agencies. This engineer will be

responsible for coordinating the development in-house of the procedures necessary to totally dismantle the facility once the fuel is shipped from site.

The Radiation Protection Engineer will be responsible for radiation protection, projections and trending. This engineer will be responsible for working with the Health and Safety Supervisor in preparing long-term prognosis for exposures and procedures necessary for decon, waste management, chemical control and fuel shipment. The Radiation Protection Engineer will assist in ensuring that an aggressive ALARA program is carried out and that contamination and background radiation exposure is reduced as low as reasonably achievable during the SAFSTOR period.

The Reactor Engineer will be responsible for all activities involving the stored fuel and will assist with plans for eventual decommissioning of the facility. This engineer will be responsible for any required reports to be generated on the stored special nuclear material.

The Safety Review Committee will remain the Offsite Review Group responsible for oversight of facility activities. It will have a quorum of 4 persons including the chairman. The Safety Review Committee will remain as currently configured (a maximum of 2 individuals having line responsibility for the plant being a part of it) but will meet at a reduced frequency. The frequency will be at least twice per year.

The Operations Review Committee (the Onsite Review Committee) will remain responsible for the review of day-to-day operations. It will consist of a quorum of at least 4 individuals drawn from the management staff at the site. It is chaired by the Superintendent. The Safety Review Committee and the Operations Review Committee will review all material as required by Technical Specifications including, but not limited to, facility changes, license amendments, and plan changes in Emergency Plan and Security Plan. The committees will also review any special tests.

6.3 CONTRACTOR USE

The use of contractors at LACBWR will continue as required throughout the SAFSTOR and DECON periods.

The use of contractors will be minimized and generally limited to areas of specialty which cannot be accomplished by Dairyland staff personnel. The use of contractors will be complementary in nature. It will highlight areas where DPC expertise or staffing is inadequate to perform specific tasks without outside help.

Contractor employment for specific tasks, possibly including monitoring or evaluating the facility during the SAFSTOR or aiding in dismantlement or cleanup during the DECON, will continue to be fully controlled by the LACBWR Quality Assurance Program.

Contractors will be selected in each case on a basis of ability, price, past performance and regulatory requirements.

The licensee, Dairyland Power Cooperative, will retain full responsibility for the performance of contractor tasks and will provide the supervision necessary to ensure that the tasks performed by contractors are in full compliance with the Quality Assurance Program, the purchase agreement and other appropriate regulations.

The use of contractors has the potential of aiding the LACBWR Decommissioning Project over the next 20+ years in certain select areas of unique expertise. The ability to maximize the benefit from contractors will be closely tied to adherence to the principles stated in the Quality Assurance Program and other DPC purchasing policies and procedures.

6.4 TRAINING PROGRAM

6.4.1 Training Program Description

6.4.1.1 LACBWR has established General Employee Training (GET) requirements for all personnel who may be assigned to perform work at LACBWR.

6.4.1.2 In addition to GET, programs have been designed to initially qualify personnel, and maintain their proficiency, in the following areas:

a) Health Physics Technician (HPT)
b) Operator
c) Certified Fuel Handler (CFH)

6.4.1.3 Special infrequently performed evolutions relating to decommissioning activities may be included for training as they approach. These evolutions may typically be:

a) Cask Handling
b) Systems Internals and Equipment Decontamination and Dismantling
c) Special Tests
d) Any other evolution determined by plant management to require special training.

6.4.2 General Employee Training (GET)

6.4.2.1 All personnel either assigned to LACBWR, or who may be assigned duties at LACBWR, will receive GET commensurate with their assignment. This training will include, as appropriate:

a) Emergency Plan Training
b) Security Plan Training
c) Radiation Protection Training
d) Quality Assurance Training
e) Respiratory Protection Training
f) Industrial Safety, First Aid, and Fire Protection

6.4.3 Technical Training

The following areas consist of a formal initial training program, followed by a recurring continuing training program.

6.4.3.1 The Health Physics Technician (HP) Initial Training Program consists of the following topics:

a) Science Training

   (1) Nuclear Theory

   (2) Chemistry

      (a) Non-radiological
      (b) Radiochemistry

   (3) Radiological Protection and Control (including surveys)

b) Systems Training

   (1) Effluent Systems Sampling and Control

c) Emergency Plan Training

   (1) Onsite Survey Team Member
   (2) Nearsite Survey Team Member
   (3) Duty HP
   (4) Re-entry Team Members
   (5) PASS Sampling
   (6) Medical Emergency

d) Environmental Program

e) Waste Disposal

f) Whole Body Counting

g) Respiratory Protection Program

h) Radiation Monitoring and Instrumentation

i) Administrative Requirements

j) First Aid Training

6.4.3.2 The Health Physics Technician (HP) Continuing Training Program consists of the following:

a) The program will be of 24-month duration, and will be repeated each 24 months.

b) Health and Safety (H&S) management will review significant industry events and distribute, as required reading to all technicians, those events determined to be applicable to LACBWR HPT's.

c) H&S management will review LACBWR events and distribute, as required reading to all technicians, those events determined to be relevant and significant.

d) Emergency Plan Training commensurate with duties.

e) Procedure changes will be reviewed by H&S management and those determined to be relevant to the performance of a technician's duties will be distributed as required reading.

f) The H&S Supervisor may initiate additional training for the technicians at any time. This training could be for, but not limited to, any of the following:

   (1) Equipment upgrade/replacement.
   (2) Infrequent and/or important tasks.
   (3) Significant procedure or department policy changes.
   (4) Significant performance problems.

g) The H&S Supervisor will ensure all Journeyman Technicians successfully complete the HP continuing training OJT cards. Satisfactory completion of the items on the Progress Cards will be denoted by a H&S management signature or initials and dated. The continuing training OJT cards wil cover the following topics.

   (1) Intralaboratory comparisons in analytical chemistry - crosscheck analysis.

   (2) Emergency Plan training.

h) A meeting will be conducted, at least semiannually, by the H&S Supervisor for all technicians for the purpose of discussing any pertinent information on the following topics:

   (1) Significant Plant/Industry events.
   (2) Equipment Changes.
   (3) Management/Technician Concerns.
   (4) Performance Problems.

Minutes of these meetings will be taken.

6.4.3.3 Operator Training Program

6.4.3.3.1 Operators assigned to LACBWR will be qualified to perform the duties of Auxiliary Operator (AO) and Control Room Watch (CRW).

6.4.3.3.2 The Operator Initial Training Program consists of the following:

a) Part I - The initial GET and Indoctrination is presented to give the new employee background information concerning the LACBWR organization, radiation safety, payroll practices, and general plant description and administration.

b) Part II - The second part of the training program, "Initial Plant Qualification Program," provides a comprehensive outline of material considered necessary for the training of individuals to qualify them for all operator duties. Periodic written and/or oral examinations, plus actual demonstrations of proficiency in practical factors, will be required of the trainees to determine their progress in the program.

c) Operator Sciences Training

(1) Nuclear Theory
(2) Radiological Protection and Control
(3) Electrical Theory, as applied to operators
(4) Chemistry

d) Operator Systems Training

(1) Plant Specific Systems
(2) Design Bases
(3) Flow Paths
(4) Components
(5) Instrumentation and Control
(6) Operational Aspects

e) Control Room Training by familiarization and manipulation under the supervision of a qualified Control Room Operator, consisting of training and exercises which apply the operating philosophy, procedures, and attitudes needed as an operator at LACBWR. The Control Room Training will be documented in the operator Practical Factors Record.

Topics include:

(1) Normal operations
(2) Malfunctions
(3) Surveillances
(4) Procedures
(5) Technical Specifications
(6) Emergency response actions

f) <u>Emergency Training</u>

(1) Emergency Plan and EPP's
(2) Plant Emergency Procedures
(3) Review of Incident Reports and LER's

g) In addition, operator trainees will take part in the LACBWR Continuing Training Program when assigned to an operating crew. This program is intended as a review for personnel and as such is not intended to serve as the sole means of training for operator trainees. All quiz and examination scores attained by trainees in the requalification program will be used to aid the trainee and not to determine his status in the program. No lecture attendance or retraining requirements are to be based on test results.

h) The candidate will normally get the necessary signatures for the Auxiliary Operator Watch Card, then Control Room Operator Watch Card and, while standing these watches, work to complete each Progress Card. As the Progress Cards are completed, the training personnel shall prepare and administer a written exam. The trainee must receive a score of $\geq$80% to pass exam.

6.4.3.4 <u>Certified Fuel Handler Training Program</u>. Under LACBWR's Operating License, NRC-Licensed Reactor Operators (RO's) and Senior Reactor Operators (SRO's) are required to manipulate the controls of the facility. Specifically, it was required that a licensed operator be present in the control room whenever fuel was in the reactor and a second RO be in the control room during reactor startups and scheduled shutdowns. Since fuel is no longer in the reactor, these requirements no longer apply.

Additionally, all core alterations were required to be performed under the supervision of a licensed SRO.

An NRC-approved requalification program in accordance with 10 CFR 55 is currently conducted to maintain the qualification of NRC-licensed operators. When the Decommissioning Plan is approved, it is planned to suspend existing NRC operator licenses.

During the SAFSTOR period, it is not expected that movements of spent reactor fuel will be made, except for special tests or inspections to monitor the

fuel in storage. At some time during the SAFSTOR period, fuel handling may be performed to transfer the spent fuel assemblies to the Department of Energy (DOE) or other entity.

A training and certification program will be implemented to maintain a staff properly trained and qualified to maintain the spent fuel, to perform any fuel movements that may be required, and to maintain LACBWR in accordance with the possession-only license. This program will provide the training, proficiency testing, and certification of fuel handling personnel. A detailed description of the Certified Fuel Handler (CFH) Program is provided in Section 10.

The Operator Training and Certification Programs ensure that people trained and qualified to operate LACBWR will be available during the SAFSTOR period. Licensee certification of personnel makes it unnecessary for the NRC to periodically conduct license examinations for persons involved in infrequent activities and prevents delays due to obtaining NRC Fuel Handler Licenses for any evolutions that may require fuel movements.

6.4.5 Other Decommissioning Training

It is anticipated that other technical topics will be presented to personnel on an as-needed basis. Current administrative guidelines will be followed to establish new procedures and to ensure the training is completed.

6.4.6 Training Program Administration and Records

The LACBWR Plant Superintendent is responsible for ensuring that the training requirements and programs are satisfactorily completed for site personnel. A LACBWR Shift Supervisor is responsible for the organization and coordination of training programs, for ensuring that records are maintained and kept up-to-date, and assisting in training material preparation and classroom instruction.

6.5 QUALITY ASSURANCE

Decommissioning and SAFSTOR activities will be performed in accordance with the NRC-approved Quality Assurance Program Description (QAPD) for LACBWR. "Safety Related" as defined would no longer be applicable in the "possession-only" mode of operation and, therefore, 10 CFR 50, Appendix "B", would no longer apply to activities performed at LACBWR.

Because of DPC's desire to maintain control and continuity in activities performed at and for LACBWR, including spent fuel and radioactive waste shipments, the QAPD will still address all 18 criteria of 10 CFR 50, Appendix "B", but some will be of a reduced scope.

A graded approach will be used to implement this program by establishing managerial and administrative controls commensurate with the complexity and/or seriousness of the activities to be undertaken.

6.6 SCHEDULE

The tentative decommissioning schedule is shown in Figure 6-2. As can be seen, DPC received a possession-only license in August 1987. This Decommissioning Plan is being submitted in December 1987. DPC expects the NRC review to be completed in mid-1989. Following approval of the Decommissioning Plan, LACBWR will enter into SAFSTOR.

As discussed in Section 7.2, some modifications are considered beneficial to support the plant in the SAFSTOR condition. Work on some of the modifications will begin in 1988, and it will be completed following approval of the Decommissioning Plan and the SAFSTOR Technical Specifications.

During the SAFSTOR period, DPC expects to ship the activated fuel to a federal repository. The timing of this action will be dependent on the availability of a federal repository and its schedule for receiving activated fuel. A modification to the Decommissioning Plan will then be submitted to describe the change in plant status and associated activities.

At this time, DPC anticipates the plant will be in SAFSTOR for a 30-50 year period. Prior to the end of the SAFSTOR period, an updated detailed DECON Plan will be submitted. The ultimate plan is to decontaminate the LACBWR facility in accordance with applicable regulations to permit unrestricted access and termination of the license.

6.7 COST ESTIMATE AND FINANCING

DPC is currently estimating a 30-50 year SAFSTOR period (Section 6.6). For cost estimating purposes, however, it was assumed that dismantlement commences as soon as possible, which would be shortly after the fuel is sent to a federal repository. The year 2011 was chosen as the earliest possible for DECON to commence. The reason for approaching financing in this manner was to assure that sufficient monies have accumulated by the earliest possible date at which they could be used. A later dismantling date will allow additional funds to accumulate to compensate for the cost of inflation. After the fuel is removed from site, SAFSTOR costs will decrease substantially. These costs will continue to be borne by Dairyland Power for the duration of the SAFSTOR period.

6.7.1 SAFSTOR (1987-2010)

The cost of SAFSTOR will be borne by the Dairyland system. The cost during the SAFSTOR period will be principally labor in scope. The cost also will include necessary administrative costs, parts and supplies and consultant support.

6. DECOMMISSIONING PROGRAM - (cont'd)

Transition year (1988)

Staff wages	$1,793,500
Fringe benefits	717,400
Employee Expenses & Training	24,956
Communication Costs	26,000
Equipment Costs	130,100
Operating & Construction Material and Expense	241,500
Maintenance Material & Expense	85,000
Outside Services including Security	1,157,293
Total for 1988	$4,175,749

Base Year (1989)

Total for 1989	$2,508,600

Total Projected for Year:

1990	$2,629,000
1991	2,755,200
1992	2,887,500
1993	3,026,000
1994	3,171,300
1995	3,323,500
1996	3,483,100
1997	3,650,300
1998	3,825,500
1999	4,009,100
2000	4,201,500
2001	4,403,200
2002	4,614,500
2003	4,836,000
2004	5,068,200
2005	5,311,400
2006	5,566,400
2007	5,833,600
2008	6,113,600
2009	6,407,000
2010	6,714,600

6.7.2 DECON (2011-2014)

The cost of deconning will be based on the selection of total radiological cleanup as the option to be pursued for the final decommissioning of the La Crosse Boiling Water Reactor. Once radioactive material and sources of contamination have been removed and the site meets established release criteria, buildings will be released for whatever activity the Cooperative

chooses to perform. They may be used for other Cooperative purposes, sold for another purpose or demolished. The cost of the DECON phase is indicative of knowledge of technology as it exists at the time of preparation of this plan (1987). It is expected that better technologies will exist by the time that this activity is carried out and Dairyland Power Cooperative is committed to the utilization of the most effective technologies available at the time in optimizing the DECON activity. The cost estimates, therefore, are stated based on today's technology in projected 2011-2014 dollars as well as in fixed 1987 dollars.

The decommissioning fund will be placed in an external fund, outside DPC's administrative control, invested in instruments such as Treasury Notes. By the end of 1987, the decommissioning fund will have accumulated to approximately $9,400,000. The decommissioning fund in the year 2000 will reach $50 million (assumed equal to $20 million in 1983 dollars, the original cost estimate). The decommissioning fund by the year 2010 will be approximately $92,600,000 accrued.

In 1983, the Dairyland Power Cooperative Board of Directors resolved to ensure adequate funding for the decommissioning of LACBWR. An annual funding of $1,300,000 was established, to be continued through 1999. This fund, with accumulated earnings, is projected to be able to adequately fund the decommissioning cost in 2010, based on the original cost estimate.

Every 5 years during the SAFSTOR period, a review of the decommissioning cost estimate will be performed in order to assure adequate funds are available at the time final decommissioning is performed. Since the cost estimate was conducted in 1983, the first update will be due in 1988.

6.8 SPECIAL NUCLEAR MATERIAL (SNM) ACCOUNTABILITY

The LACBWR Accountability Representative is the person responsible for the custodial control of all SNM located at the LACBWR site and for the accounting of these materials. He is appointed in writing by the Dairyland Power Cooperative General Manager.

The LACBWR Spent Fuel (333 assemblies) is stored under water in the high density spent fuel storage racks in the LACBWR Fuel Storage Well which is located adjacent to the reactor in the LACBWR containment building. Fifty-two new fuel assemblies will be stored in the locked new fuel storage cabinets located in the Containment Building in the reactor biological shield until they can be shipped offsite for recovery of the contained material. This unused fuel will probably be shipped offsite by mid-1988. Additional small quantities of SNM are contained in neutron and calibration sources and in fission detectors which are appropriately stored at various locations in the LACBWR plant.

All fuel handling and all shipment and receipt of SNM is accomplished according to approved written procedures. Appropriate accounting records will be maintained and appropriate inventories, reports and documentation

will be accomplished by or under the direction of the LACBWR Accountability Representative in accordance with the requirements set forth in 10 CFR 70, 10 CFR 73 and 10 CFR 74.

6.9 SAFSTOR FIRE PROTECTION PROGRAM

6.9.1 Program Administration

6.9.1.1 The LACBWR Plant Superintendent is responsible for the fire protection program. A member of the Dairyland technical staff is responsible for annual evaluation of equipment provided for fire fighting, training, and maintaining a current and effective fire protection program.

6.9.1.2 The training program for the Fire Response Team will be maintained under the direction of a designated staff member and will meet or exceed the requirements of Section 27 of the NFPA Code 1976.

6.9.1.3 The Fire Response Team will consist of 3 members. These individuals will be available to respond in the event of a fire emergency at the LACBWR Unit. The Fire Response Team Leader will be a member of the Operations Department. The Fire Response Team will not include any personnel required for other essential functions during a fire emergency.

6.9.1.4 Implementing procedures for surveillance testing and inspection, to assure that necessary equipment is in place and operable, have been established. Fire drills, conducted under the direction of the Fire Marshall, will be held for each operating shift each quarter.

6.9.1.5 Self-contained breathing apparatus will be supplied for each member of the Fire Response Team and for any control room personnel. One hour of breathing air spare bottles for each of the above required masks will be available within the confines of the unit with cascade recharging facilities located on the Genoa site.

6.9.1.6 A section of technical specifications shall delineate inspection and surveillance test frequency, reports necessary, and statements of actions.

6.9.2 SAFSTOR Analysis

LACBWR can safely maintain and control the FESW in the case of the worst postulated fire in each fire area of the plant.

The SAFSTOR fire protection program and systems were reviewed using the criteria and guidelines of Branch Technical Position 9.5-1 for general guidance. It was concluded that the LACBWR Fire Protection Program and detection and extinguishing systems are adequate, considering the reduced risk due to plant being in the SAFSTOR mode. The installed fire protection equipment being maintained during the SAFSTOR period is the same as that used during plant operation.

Fire protection practices include isolation of fire areas via sealed penetrations; detection of potential fires and location identification for the plant operators, coverage by automatic extinguishing systems in the plant, protecting cables with fire resistant coverings, and installed emergency lighting systems.

6.9.3 Plant Fire Layout

The LACBWR plant is divided into fire areas. These areas are separated from each other by one or more of the following:

1) 3-hour or better fire walls.

2) Walls and ceilings with ratings well in excess of the combustibles involved.

6.9.4 Fire Protection Systems

The LACBWR Fire Protection System and equipment provide the means to quickly combat all types of fires that might occur at the plant and to maintain the plant in a safe condition. The Fire Protection System consists of a CO_2 flooding system for the 1B Emergency Diesel Generator (EDG), Halon flooding system, portable extinguishers, sprinkler systems, hose stations and fire hydrants, transformers, deluge sprinklers, portable smoke ejectors, and a fire and smoke detection system.

6.9.4.1 Fire Suppression Water. The fire suppression water system is a combined usage water system and is called the High Pressure Service Water System (HPSW). Water is supplied from the Mississippi River which is the west boundary of the plant. A reinforced concrete flume juts out from the cribhouse to channel water to the pumps.

Two 125 psi net head, vertical turbine, diesel fire pumps are connected in parallel and take suction from the well supplied by the flume.

The operating pump discharges into a six-inch steel underground main system that loops the plant. One leg of the loop is run overhead through the grade floor of the Turbine Building at the west end. Two underground sectionalizing valves with locked curb boxes are provided to isolate the overhead main section in case of a rupture of either inside or outside buried mains. Two locked valves are also provided inside the plant for this purpose.

All HPSW services are fed from the overhead main except for five 6-inch fire hydrants which are spaced at approximately 200 feet intervals around the plant. Pressure is maintained at 80 to 120 psi in the HPSW system by a 500 gpm electric, fire booster pump in the west end of the grade floor of the Turbine Building. This pump takes suction from the 16-inch Low Pressure Service Water (LPSW) main. LPSW is supplied by one of two large electric vertical turbine pumps in the Cribhouse. The HPSW system inside of the building can be isolated from the outside underground loop in case of an underground break.

As a backup system to the HPSW system, the LACBWR fire main can be cross connected to the Genoa-3 station fire main.

6.9.4.2 Sprinkler Systems. There are seven sprinkler systems located throughout the plant. Five of these are of the fusible link type, one is a dry pipe sprinkler requiring manual initiation of flow, and one system is initiated by either automatic or manual means.

Four of the fusible link sprinklers are located above the turbine oil tank, above the Alternate Core Spray ac and dc valves, above the HPSW diesels in the cribhouse, and in the heating oil storage tank room. All of these systems have flow switches installed in their supply piping with associated alarms at the control room fire panel.

The dry pipe sprinkler system is located in the electrical penetration room.

The other system is a transformer deluge system in the LACBWR switchyard.

6.9.4.3 Fire Hose Protection. Outside hose is jacketed, lined, 300 psi test fire hose with the necessary approved fog nozzles and fittings. This equipment is stored in four strategically located cabinets. The hoses are adequate to provide at least two 1.5 inch streams on any fire in the plant. There is also hose available in the outside cabinets to cross connect LACBWR with Genoa-3 plant if necessary.

Inside hose cabinets are provided at five locations in the Turbine Building. Each cabinet contains 75 feet of approved, 1.5 inches, lined hose and an approved fog nozzle. A similar cabinet is located in the Waste Treatment Building.

A hose reel with approved 1.5 inch, jacketed and lined, fire hose and approved fog nozzle is installed in the 1B Diesel Generator electrical equipment room.

The Containment Building has four hose reels, each with 75 feet of 1.5 inch rubber hose and adjustable nozzle. These hose connections are at the four main levels of the building and within easy reach of any fire that might occur on those levels.

All hose fittings, nozzles, hydrant butts and hose couplings are National Standard Fire Hose Thread.

6.9.4.4 Special Hazard Extinguishing Systems. The special hazard extinguishing systems at LACBWR are the approved high pressure carbon dioxide total flooding system for the 1B Diesel Generator room and Halon systems for the electrical equipment room and the record storage room, which is in the administration building.

Due to the diminished probability of a health and safety threat with the reactor non-operable, the need for Halon protection in the electrical equipment room has also diminished and will not be needed. This area is alarmed and can be treated as a Class A fire area.

6.9.4.5 Alarm and Detection System. A Class B smoke and fire alarm system has been installed throughout the plant. The system has an alarm panel in the main control room. Alarms connected to the system include smoke and fire detectors and sprinkler alarms. Fire pump, CO_2, and Halon systems also alarm in the control room.

Areas of potential fires throughout the plant are covered by approved, ionization chamber-type detectors and/or thermal fire detectors which combine rate-of-rise and fixed temperatures, which are connected to 20 separate alarm zones on the fire alarm panel. The pilot light on the detector indicates which detector in an alarm zone has been activated.

A fire alarm zone bypass switch panel is located in the control room, next to the multi-zone fire cabinet. In the "BYPASS" position, the alarm circuit for that zone is bypassed, the 1-1/2 hour timer motor for that zone is started and the red flashing light at the zone detector head is turned off if the detector was alarming. This allows welding or other maintenance to be conducted in a zone without disabling the entire fire detection system.

Four of the sprinkler systems are monitored by flow alarms to indicate a water flow in the systems. The alarm is initiated by a water flow switch in each of the systems and is alarmed at the fire alarm panel.

The two diesel fire pumps are alarmed to an alarm panel in the control room showing trouble and operating alarms.

6.9.4.6 Inspection Services. The diesel fire pumps are tested on a monthly frequency and the data recorded. In addition, an annual test of the flow characteristics of the pumps and piping system is performed for the property insurance carrier.

Various other tests and inspections of all equipment are made by plant personnel and the fire insurance company retained by Dairyland Power Cooperative.

6.9.5 Fire Fighting Equipment

6.9.5.1 Fire Extinguishers. An adequate supply of portable extinguishers of appropriate types are provided throughout the plant.

6.9.5.2 Outside Storage Cabinets and Equipment. Four outside cabinets have been provided for storage of equipment necessary for use of the fire hydrants. One cabinet is located on the northeast corner of the Waste Treatment Building, one east of the Cribhouse, one by 1B Diesel Building, and one at the southwest corner of the Turbine Building. All are painted

vermillion red for easy identification. Equipment consists of: 2-1/2" hose, 1-1/2" hose, fog nozzles, gate valves, hydrant wrench, hose spanner wrenches, and coupling gaskets.

6.9.5.3 Tool Kits. Tool kits have been provided in two locations, one located in an outside hose cabinet, the other located in the maintenance shop on the north wall.

6.9.5.4 Self-Contained Breathing Apparatus. Self-contained breathing apparatus is located in the turbine hall outside the control room door and in the change room.

6.9.5.5 Portable Smoke Ejectors. Smoke ejectors are provided for the removal of smoke from confined or non-ventilated areas. These are stored along the north stairs to the machine shop and are of 2000 cfm rating.

6.9.5.6 Firefighting Clothing and Equipment. Fire clothing and equipment is stored in the change room. There are enough coats, boots, gloves, and fire fighter helmets for the Fire Response Team.

6.10 SECURITY DURING SAFSTOR AND/OR DECOMMISSIONING

During the SAFSTOR status associated with the LACBWR facility, security will be maintained at a level commensurate with the need to insure safety is provided to the public from unreasonable risks.

Guidance and control for security program implementation are found within the LACBWR Security, Safeguards Contingency, and Guard Force Training and Qualification Plans, along with the Security Control Procedures.

6.11 RECORDS

Any records which are generated for the safe and effective decommissioning of LACBWR will be placed in a file explicitly designated as the decommissioning file. Any records generated which are not specifically for decommissioning, but could affect decommissioning activities, will be indexed, filed, and retrieved in accordance with the LACBWR Quality Assurance Program Description (QAPD).

Examples of records which would be required to be placed in the decommissioning file are:

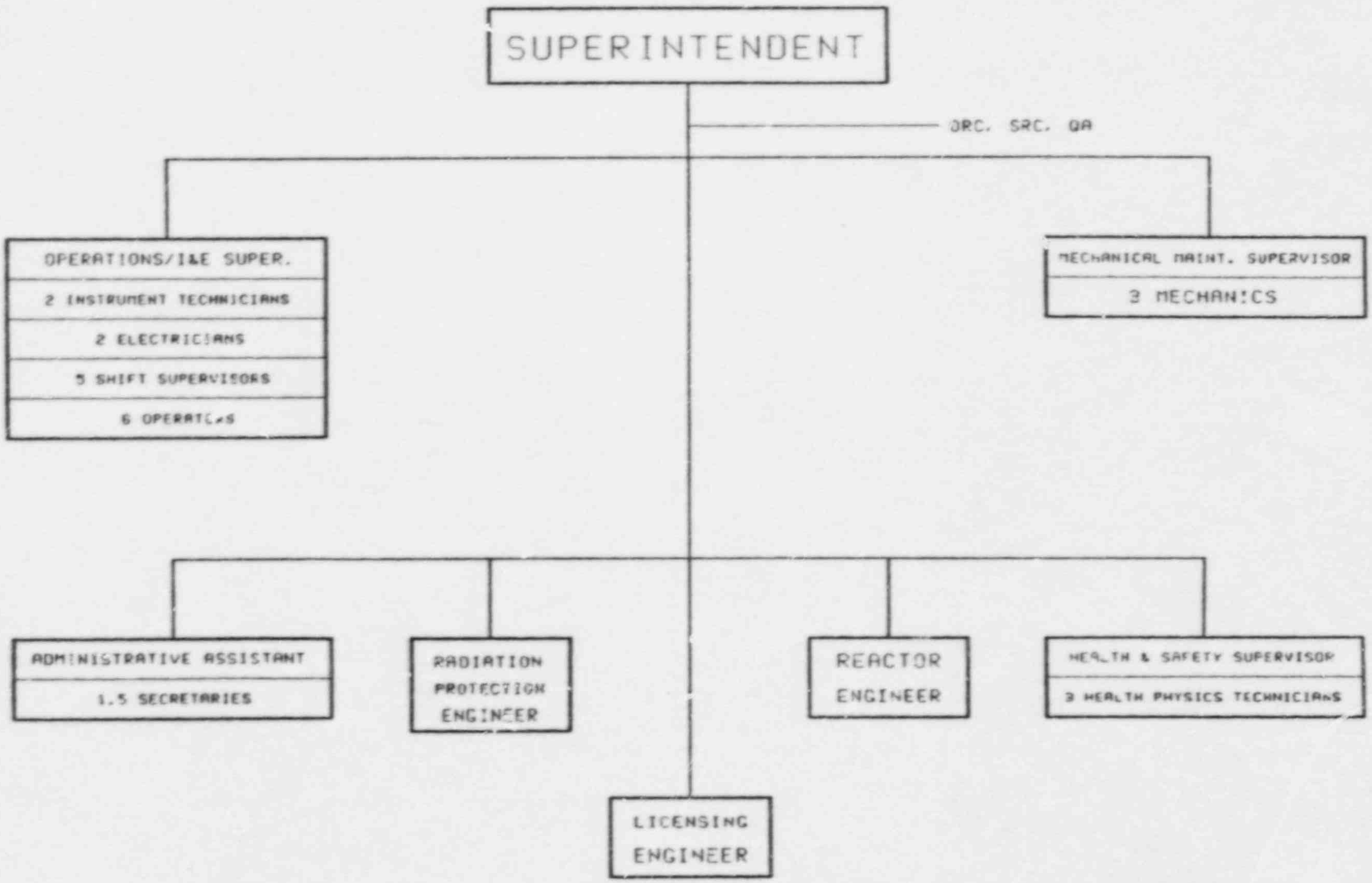
- Records of spills or spread of radioactive contamination, if residual contamination remains after cleanup.
- Records of contamination remaining in inaccessible areas.
- Plans for decontamination (including processing and disposal of wastes generated).

6. DECOMMISSIONING PROGRAM - (cont'd)

- Base line surveys performed in and around the LACBWR facility.
- Analysis and evaluations of total radioactivity concentrations at the LACBWR facility.
- Any other records or documents, which would be needed to facilitate decontamination and dismantlement of the LACBWR facility and are not controlled by other means.

Supporting documentation (i.e., as-built drawings, modifications, procedures, etc.) will be generated, maintained, updated, filed, and controlled in accordance with the LACBWR-approved QAPD.

LA CROSSE BOILING WATER REACTOR SAFSTOR STAFF



ASSUMES COOPERATIVE-WIDE SECURITY & QA

FIGURE 6.1

Activities	Year / Period	1987 2nd Qtr	1987 3rd Qtr	1987 4th Qtr	1988	1989	1990 - 2039
Reactor Shutdown		x					
File for Possession-Only License		x					
Reactor Defueling		x					
Receive Possession-Only License			x				
File Technical Specifications for Interim Period		————	————				
Submit Decommissioning Plan				x			
Submit SAFSTOR Technical Specifications				x			
Perform Baseline Radiation Survey			————				
Perform System Modifications					————————	————————	————
Decommissioning Plan Approval						x	
*SAFSTOR Period						————	———————— - - - - - - - -
**Shipment of Fuel Offsite							x
**Modification to Decommissioning Plan for SAFSTOR							x
*Update DECON Plan							- - - - - - - - - -
*Commence DECON							- - - - - - - - - -

* SAFSTOR period is expected to last 30-50 years. A detailed DECON Plan will be submitted prior to end of SAFSTOR period.

** Dependent on schedule of federal repository.

Tentative Schedule for LACBWR Decommissioning

FIGURE 6.2

7. DECOMMISSIONING ACTIVITIES

7.1 PREPARATION FOR SAFSTOR

The plant was shut down on April 30, 1987. Reactor defueling was completed June 11, 1987. Since the plant shut down, some systems have been secured. Additional systems will be shut down following determination of layup methodology. Others are awaiting changes to plant Technical Specifications. Section 5.2 discussed the plant systems and their status.

One of the preparations for SAFSTOR is removal of the unused reactor fuel from the site. Options are currently being evaluated for disposition of the 52 new fuel assemblies.

Some modifications are planned for the maintenance of the plant in the SAFSTOR condition. These modifications are discussed in Section 7.2. Some may be performed, in accordance with plant modification procedures, which require a 50.59 review, during the interim period while the Decommissioning Plan is being reviewed, prior to entry into SAFSTOR.

In addition to preparation of this Decommissioning Plan, proposed revisions to Technical Specifications, the Security Plan, the Emergency Plan, and the Quality Assurance Program Description have been completed. An addendum to the Environmental Report and a preliminary DECON plan are also being submitted. As changes to these documents are approved, action will be taken to meet the revised requirements.

7.2 PLANNED MODIFICATIONS

The LACBWR staff reviewed the facility to determine if any modifications should be implemented to enhance safety or improve monitoring during the SAFSTOR period while fuel is stored onsite. Some modifications were evaluated as being beneficial and so will be performed.

The majority deal with the Fuel Element Storage Well System (FESW). A redundant FESW level indicator will be added. A second remote manually-operated FESW makeup line will be installed, which will supply water from the Overhead Storage Tank. The FESW cleanup system will be improved. A thermocouple will be placed in the storage well, which will read out in the control room. Also, an improved local direct means of measuring FESW water level will be installed.

Even though credit is not taken in the safety analyses (Section 9) for containment integrity, the automatic closure signals for containment isolation valves which will still be used will be modified. The valves will close on either a high Containment Building activity signal or a low FESW level signal, which will be set below the normal water level range. An FESW level indicator will be used to generate the low FESW level signal. A new Containment Building activity monitor is planned, which would generate the high activity signal.

In addition to replacing the existing Containment Building activity monitor, plans exist to recalibrate all gas activity monitors. The monitors will be recalibrated using Kr-85, which is the isotope most appropriate for the SAFSTOR period.

As discussed in Section 7.1, some of these modifications may be made in the interim period before entering SAFSTOR. Implementation of some modifications will take place after the Decommissioning Plan is approved. Approval of revised Technical Specifications is necessary in some cases.

7.3 ACTIVITIES DURING SAFSTOR PERIOD

7.3.1 Flushing Systems and Decontamination During SAFSTOR

During the SAFSTOR period, selected systems and components, especially those in accessible areas, will be flushed or decontaminated. Surface areas in accessible areas will continue to be decontaminated. The principal reasons for a selected flushing and decontamination program are:

1) To reduce the contamination levels and radiation dose rates in areas that will be accessible for periodic maintenance and surveillance activities during the SAFSTOR period.

2) To reduce radiological surveillance requirements.

3) To reduce the need for protective equipment for personnel conducting maintenance and surveillance.

4) To reduce the inventory of radioactive material and the potential for the transfer of radioactive material to non-controlled areas.

7.3.1.1 Internal System Flushing. Various closed plant systems, which contain water, will be flushed by recirculating water through the system's piping, vessels, tanks and other components, with subsequent removal of suspended solids and radioactive ions by filtration and/or demineralization. Some of these systems or components will be drained after the flushing operations indicate that further reductions in radioactivity by this method are impractical. These systems may be maintained in a dry layup condition. Other systems will not be drained and may be maintained in a wet layup condition to reduce radiation dose rates in accessible locations. In some cases, the installation of additional shielding to reduce radiation dose rates near the accessible areas of previously flushed systems or components may be necessary for ALARA.

7.3.1.2 Area and System Decontamination. The decontamination program during the SAFSTOR period will be a continuation of routine decontamination work performed at LACBWR. Plant areas and component outer surfaces will be decontaminated to reduce the requirements for protective equipment use and to reduce the potential for the translocation of radioactive material. Decontamination methods that are used are dependent upon a number of variables,

such as surface texture, material type, contamination levels, and the tenacity with which the radioactive material clings to the contaminated surfaces.

Surface areas are primarily decontaminated using hand wiping, wet mopping, and wet vacuuming techniques. Detergents and other mild chemicals may be used with any of these techniques. The residual water cleaning solutions are collected by floor drains and processed through the liquid waste system. Most areas are routinely decontaminated to levels below 2000 dpm/ft² (about 500 dpm/100 cm²). Many areas are maintained below the Lower Limit of Detection (LLD). Efforts will be made to maintain all accessible areas in the plant as free of surface contamination as is reasonably achievable.

Small tools and components will be periodically decontaminated by wiping with cleaning agents, steam cleaning, abrasive blasting, dishwasher, ultrasonic cleaning, electropolishing or other methods. Some unused equipment may be decontaminated as a prior step to removal for disposal as commercial or radioactive solid waste. Some unused equipment may be decontaminated prior to continued use in unrestricted areas.

Larger systems and components in accessible areas may be decontaminated using hydrolazers, abrasives, chemicals or other methods, after appropriate ALARA and economic evaluations are conducted.

7.3.2 Removal of Unused Equipment During SAFSTOR

During the SAFSTOR period, some equipment and plant components will no longer be considered useful or necessary to maintain the plant in the SAFSTOR condition. Some equipment located in non-controlled areas may be transferred directly for use at another location or disposed of as commercial solid waste.

Some unused equipment or components located within controlled areas, which have not previously been used for applications involving radioactive materials will be thoroughly surveyed and documented as having no detectable radioactive material (less than LLD) prior to transfer to another user or disposal as commercial solid waste.

Other unused equipment or plant system components which have previously been used for applications involving radioactive materials may be removed, thoroughly surveyed and transferred to another licensed user, or disposed of as low level solid radioactive waste material. Some equipment may be decontaminated and will be surveyed to verify that it contains no detectable radioactive material (less than LLD), prior to transfer to an unlicensed user, or for disposal as commercial solid waste.

Removal of plant equipment wil be performed only after review. A 50.59 safety evaluation will be conducted prior to dismantling any system.

Asbestos removed from plant systems will be handled in accordance with the Dairyland Power Cooperative asbestos control program.

7.3.3 Research

During the SAFSTOR period, an Aging Research Program may be conducted. This program may entail records research and possible removal of unused components for testing.

7.3.4 Testing and Maintenance Program to Maintain Systems in Use

During the SAFSTOR period, a testing and maintenance program will continue for those systems previously designated as being required for SAFSTOR. Routine preventive maintenance will be performed as before, but where the present maintenance interval is listed as "Outage," a new interval will be specified. Corrective maintenance will be performed as necessary. Instrument calibrations and other routine testing will continue as before for equipment which will be required to be operable.

7.4 PLANT MONITORING PROGRAM

Activities and plant conditions at LACBWR will continue to be maintained to protect the health and safety of both the public and plant workers. Baseline radiation surveys have been performed to establish the initial radiological conditions at LACBWR during SAFSTOR. An in-plant as well as offsite surveillance program will be established and maintained to assure plant conditions are not deteriorating and environmental effects of the site are negligible.

7.4.1 Baseline Radiation Surveys

Baseline surveys have been performed to establish activity levels and nuclide concentrations throughout the plant and surrounding area. These surveys included:

a) Specific area dose rates and contamination levels.

b) Specified system piping and component contact dose rate.

c) Radionuclide inventory in specified plant systems.

d) Radionuclide concentration in the soil and sediment in close proximity of the plant.

Baseline conditions will be compared with routine monitoring values to determine the plant/system trends during SAFSTOR. Some specific monitoring points may be reassigned during the SAFSTOR period if it is determined that a better characterization can be obtained based on radiation levels measured or due to decontamination or other activities which are conducted and experience achieved.

7. DECOMMISSIONING ACTIVITIES - (cont'd)

7.4.2 In-Plant Monitoring

Routine radiation dose rate and contamination surveys will be taken of plant areas along with more specific surveys needed to support maintenance at the site. A pre-established location contact dose rate survey will be routinely performed to assist in plant radionuclide trending. These points are located throughout the plant on systems that contained radioactive liquid/gases during plant operation.

7.4.3 Release Point/Effluent Monitoring

During the SAFSTOR period, effluent release points for radionuclides will be monitored during all periods of potential discharge, as in the past. The two potential discharge points are the stack and the liquid waste line.

a) Stack - the effluents of the stack will be continuously monitored for particulate and gaseous activity. The noble gas detector(s) will be recalibrated to Kr-85. The stack monitor will be capable of detecting the maximum Kr-85 concentration postulated from any accident during the SAFSTOR period. Filters for this monitor will be changed and analyzed for radionuclides on a routine basis established by plant Technical Specifications.

b) Liquid discharge - the liquid effluents will be monitored during time of release. Each batch release will be gamma analyzed before discharge to ensure plant Technical Specifications will not be exceeded.

All data collected concerning effluent releases will be maintained and will be included in the effluent report.

7.4.4 Environmental Monitoring

Offsite area dose rates as well as fish, air, liquid, and earth samples will continue to be taken and analyzed to ensure the plant is not adversely affecting the surrounding environment during SAFSTOR. The necessary samples and sample frequencies will be specified in plant Technical Specifications. Meteorological data will be continuously collected to be used in offsite dose calculations to ensure the safety of the general public.

All data collected will be submitted in the annual environmental report.

8. HEALTH PHYSICS

During the SAFSTOR period of LACBWR, radiation protection and health physics programs will be provided to ensure the health and safety of LACBWR workers. The programs will also provide the necessary monitoring and control of radiological conditions to protect the health and safety of the general public and to ensure compliance with LACBWR license requirements. In addition, programs will be provided to maintain radiation exposures as low as reasonably achievable (ALARA).

8.1 ORGANIZATION AND RESPONSIBILITIES

The organization described below is the organization as it is expected to exist during the SAFSTOR activities. The organization may be changed slightly during the SAFSTOR period as staffing levels requirements change. Responsibilities assigned to a position which is deleted will be assigned to another individual in order to maintain continuity.

The LACBWR Plant Superintendent has the overall responsibility for all onsite activities including assurance that ALARA policies and the radiation protection program are carried out. He is the chairman of the ORC. He is also responsible for approving all plant procedures.

Health Physics management shall provide the first-line supervision, training and technical assistance to the Health Physics department. Management personnel will report directly to the Plant Superintendent. They shall assure that all ALARA policies and all aspects of the Radiation Protection Program are implemented. They shall also be members of the ORC. Health Physics management will be responsible for all departmental budgeting and scheduling.

The Health Physics Technician (HP) will perform chemical and radiological sampling, surveys and analysis as directed by HP management. In addition, they will also be responsible for conducting the personnel monitoring program, maintaining radiation protection records and monitoring work in progress within the radiologically restricted area.

8.2 ALARA PROGRAM

8.2.1 Basic Philosophy

The radiation exposure criteria set forth shall be for the protection of personnel against radiation hazards arising from work associated with LACBWR. They require that no person under 18 years of age shall be occupationally exposed to ionizing radiation. A continuous effort should be made to reduce levels of radiation and radioactivity in order to maintain radiation doses at the lowest achievable value below the established limits of 10 CFR 20.101.

A further goal of the Health Physics Procedures in use at LACBWR shall be to reduce personnel exposures to radiation and radioactive material to As Low As Reasonably Achievable (ALARA).

8.2.2 Application of ALARA

a) To obtain the goal of ALARA, the TOTAL dose to be received during a specific job and the total allowable for the year for the entire operation of the facility should be balanced.

b) The dose received by an individual from occupational exposures will be considered with respect to his/her daily, quarterly, yearly, lifetime, internal, and external accumulation. The individual's TOTAL dose should be balanced with the TOTAL dose received by the entire LACBWR work force and temporary DPC/contractor employees to aid the overall ALARA Program.

c) A Special Work Permit (SWP) ALARA review should be conducted if a job is expected to require between 2.0 and 5.0 manRem. An SWP ALARA review form used for this application should be governed by total manRem estimates, based upon current surveys and job-time estimates, and total manRem for past similar jobs based upon an SWP dose accountability file.

d) An SWP ALARA review will be conducted if a job is expected to require between 5.0 and 10.0 manRem.

e) If a job is expected to require greater than 10.0 manRem, a more intensive ORC ALARA review will be required. This may include the use of special procedures.

f) Documentation of ALARA engineering work and cost benefits will be maintained in files.

g) HP management will conduct a semi-annual ALARA review of Actual versus Projected (goal) exposures. ManRem exposures will be reviewed regularly with the Plant Superintendent. Included should be a review of the effectiveness of specific steps that were taken to reduce radiation exposure (ALARA Engineering).

8.2.3 Radiation Exposure Limits

a) Daily Administrative Limit

An administrative guideline of 100 mRem per day will not be exceeded without the prior approval of the Health and Safety Supervisor or alternate. Assignment to an SWP by the Health and Safety Supervisor (or his authorized representative) will authorize an individual to exceed the 100 mRem per day administrative limit.

b) Quarterly Administrative Limit

Administratively, personnel will be limited to 2.7 Rem per calendar quarter for external, whole body penetrating radiation. If personnel are expected to exceed 1.25 Rem in a calendar quarter, they will be placed on an Alert List. Additionally, if a person has exceeded 1.25 Rem in a calendar quarter and is expected to exceed 2.0 Rem in that calendar quarter, the person will have written justification and approval to receive an increase in the allowable dose increment from 2.0 Rem to 2.7 Rem.

c) Yearly Administrative Limit

Administratively, personnel will be limited to 5 Rem per calendar year of external/whole body penetrating radiation. Every effort will be made to equalize exposures of all personnel in accordance with ALARA principles. If personnel are expected to exceed 5 Rem in a calendar year, written approval to exceed the 5 Rem guideline will be completed by the Plant Superintendent prior to the person exceeding 5 Rem.

8.3 RADIATION PROTECTION PROGRAM

The radiation protection program that will be utilized during the SAFSTOR period will be an extension of the program that was used during the period of reactor operations at LACBWR. This program is in compliance with the requirements of 10 CFR 20. Implementation of the radiation protection program will be done at LACBWR through Health and Safety procedures. The following section describes the radiation protection program.

8.3.1 Personnel Monitoring

To ensure that the radiation exposure limits of 10 CFR 20 are not exceeded, a personnel radiation exposure monitoring system will be maintained. Two basic means shall be used to evaluate each individual's radiation exposure:

a) TLD/Film Badges - to give integrated dose measurements over relatively long periods of time.

b) Self-Reading Dosimeters - to give interim indication of accumulated doses.

Film badges and self-reading dosimeters will be worn by all plant personnel entering the radiological restricted area. They will be worn at or above the waist and on the front of the body, unless the Health and Safety management specifies that the badges be worn differently. Extremity dosimetry will be worn by all personnel when conditions exist that could cause a significantly higher than whole body dose to be received by a worker's extremities.

Long-term visitors expecting to receive a significant radiation exposure (300 mRem whole-body, 4500 mRem extremity, or 1850 mRem of skin exposure), persons entering a high radiation area, or a person under 18 years of age who is likely to receive greater than, or equal to, 60 mRem of whole-body, 900 mRem of extremity or 350 mRem of skin exposure will be issued TLD/film badges and dosimeters and will be monitored in the same manner as the regular plant personnel.

Casual and short-term visitors (those for whom exposures are expected to be insignificant) will be issued pocket dosimeters only.

Monthly film badge records received from the badge processor will be evaluated and maintained. Periodic quality testing of TLD/film badges and pocket dosimeters will be conducted.

The LACBWR whole body counter will be used to detect any internal contamination for:

a) All new employees who will routinely work with radioactive material.

b) Any individual suspected of having received any internal deposition.

c) Routinely (annually for all plant personnel).

d) Upon termination of any employee who worked with radioactive material.

If it is determined that any employee has a significant internal deposition of any isotope, he may be required to submit a urine and/or fecal specimen as directed by LACBWR Health and Safety.

All personnel leaving a controlled area will be required to conduct a personnel contamination survey using the contamination detection instrument provided at the exit.

8.3.2 Respiratory Protection Program

A respiratory protection program will be maintained during the SAFSTOR period.

The Health and Safety Supervisor is responsible for the Respiratory Program at LACBWR. The Health and Safety Supervisor or designated alternate will evaluate the total job hazard, recommend engineering controls if appropriate, specify respiratory protection if control cannot be otherwise obtained and forbid the use of respirators if conditions warrant. The Health and Safety Department is responsible for the selection, care, and maintenance of all respiratory protection equipment that falls under the scope of the respiratory protection program.

The acceptable manner for preventing the internal exposure of personnel from radioactive airborne contamination is to control radioactivity concentration in the air breathing zones. Whenever possible, this will be accomplished by the application of engineering control measures such as containment, decontamination, special ventilation equipment and design. The use of personal respiratory protective equipment as a primary control is undesirable and is acceptable only on a non-routine basis or in an emergency situation.

Equipment such as hoods, blowers, and filtered exhaust systems will be used to provide controls for routine operations and, whenever possible, for non-routine operations. In some cases, such controls may be inadequate or impractical and the use of protective breathing apparatus will be approved on a short-term basis.

The periods of time for which respirators may be worn continuously, and the overall time of uses, should be kept to a minimum. The wearer shall leave the area for relief from respirator use in case of equipment malfunction, undue physical or psychological discomfort, or any other condition that, in the opinion of the user, his supervisor or the Health and Safety Department, might cause significant reduction in the protection afforded the user.

Respiratory protection equipment will be issued to individuals only after documentation has been received that shows that the person has satisfactorily completed:

a) medical exam,

b) respiratory protection training, and

c) respiratory fit test (does not apply to in-line supplied air hoods and Self-Contained Breathing Apparatus).

8.3.3 Protective Clothing

Personnel working in contaminated areas of LACBWR are provided with protective clothing to minimize the potential for personnel contamination. Routine entry into a contaminated area will require a minimum protective clothing requirement of:

a) coveralls
b) head covering
c) gloves
d) shoe coverings

Specific jobs may require additional protective clothing. These additional requirements will be determined by the Health and Safety Department and will be listed on the Special Work Permit for the job.

During the SAFSTOR period, the laundry facility will remain operational to ensure an adequate supply of clean protective clothing.

8.3.4 Access Control

To limit radiation exposures, personnel access is controlled in areas where such exposure is possible. This control consists of a system of physical barriers, warning signs and signals.

A Special Work Permit (SWP) will be issued as authorization for personnel to perform work of a non-routine nature in a specific area which involves unusual hazards. SWP's will be used to inform personnel of these hazards and the safeguards/protective measures which need to be taken during the work to ensure their well being.

8.3.5 Postings

Postings shall be in accordance with the requirements of 10 CFR 20.203 and 10 CFR 20.204, as applicable.

8.4 RADIATION MONITORING

A program for routine surveys and monitoring will be continued during the SAFSTOR period at LACBWR. This program will continue to assure all personnel are aware of the possible hazards involved before entering a potential radiation area or a potentially contaminated area. This will be done to ensure that the potential hazards are adequately defined, that adequate controls are instituted so that radiation exposure to personnel working in radiation areas or working with radioactive materials is minimized, and that each person carries out his work in a radiologically safe manner.

Survey data records will be maintained to assist in the evaluation of the radiological conditions and trends at LACBWR during SAFSTOR activities.

The radiological monitoring program will include the following surveys:

a) airborne activity surveys
b) dose rate surveys
c) contamination surveys
d) liquid activity surveys
e) environmental surveys

8.4.1 Airborne Radioactivity Surveys

In addition to using the fixed location or mobile air monitors, particulate airborne activity shall also be determined as needed by drawing a sufficient quantity of air through a filter paper. The samples shall be counted for beta-gamma activity in gas-flow proportional detector and scaler equipment. Alpha activity of a sample shall be determined by means of a windowless gas-flow proportional detector and a scaler when alpha radioactivity is suspected of being present. Samples are analyzed for specific isotopic concentrations, by the use of a gamma analyzer. Particulate samples of the stack releases will be obtained and analyzed weekly to determine release rates.

Non-routine air samples to establish protection requirements for maintenance activities or to verify airborne radioactivity conditions during work activities are obtained and analyzed when routine samples are not sufficient for monitoring plant conditions.

8.4.2 Radiation Surveys

Radiation surveys are conducted for the following purposes:

a) Measure and document radiation and contamination levels in areas of interest.

b) Identify trends in radiation and contamination levels, particularly during work in progress.

c) Determine appropriate protective measures for personnel working in controlled areas.

d) Provide information so that workers can maintain their doses ALARA.

e) Identify locations and situations where special dosimetry is required.

In addition to the measurements made by the fixed-location area radiation monitors, the measurement of external dose-rates shall be accomplished by portable survey instruments. The operation of the survey instruments shall be in accordance with the operating instructions outlined in each particular instrument manual or by procedure. Instruments covering high, intermediate, and low ranges shall be available on site.

Surveys will be conducted by the Health and Safety Department to determine general area dose rates. They will also monitor areas to locate any radiological hot spots. Surveys will be performed on a routine basis established by procedures.

Special radiation surveys of particular items or areas are performed on an "as needed" basis. Examples of special radiation surveys are the removal of equipment or materials from a controlled area, leak testing of sealed radioactive sources, or the shipment or receipt of radioactive material packages.

8.4.3 Contamination Surveys

Contamination surveys will be conducted routinely by the Health and Safety Department as established by procedure to determine area contamination levels.

Special contamination surveys of particular items or areas are performed on an "as needed" basis. Examples of special contamination surveys are the removal of equipment or materials from a controlled area, leak testing of sealed radioactive sources, or the shipment or receipt of radioactive material packages.

A dry filter paper or cloth disc will be wiped over approximately one square foot (12"x12" square or 12'-long S-shaped) of the surface being monitored. Swipes will be counted for beta-gamma activity in a gas-flow proportional detector or with a 2 π GM probe or equivalent in fixed geometry sample holder as necessary. Alpha activity of a swipe will be determined by means of a windowless gas-flow proportional detector and a scaler or equivalent, when alpha radioactivity is suspected of being present.

8.4.4 Liquid Activity Surveys

Samples of water containing radioactivity are collected and analyzed on a routine basis. Spent fuel pool water is analyzed to detect indications of degradation of the fuel stored in the pool. Samples of liquid radioactive wastes and processed wastes are analyzed to ensure levels of radioactivity are below the levels permitted for release. Samples are analyzed by Health and Safety Department personnel in accordance with established procedures.

8.4.5 Environmental Surveys

Environmental samples will be taken within the surrounding areas of the plant. These samples will be analyzed to determine any effects plant effluent releases may have on the environment. This program will be conducted to comply with LACBWR Technical Specifications.

8.5 RADIATION PROTECTION EQUIPMENT AND INSTRUMENTATION

A variety of equipment and instruments are used as part of the radiation protection program. Equipment and instrumentation are selected to perform a particular function. Sensitivity, ease of operation and maintenance, and reliability are factors that are considered in the selection of a particular instrument. As the technology of radiation detection instrumentation improves, new instruments are obtained to more accurately measure radioactivity and ensure an effective radiation protection program.

This equipment can be broken down into several specific groups each with its own dedicated functions. These groups are:

a) Portable Instruments
b) Installed Instruments
c) Personnel Monitoring Instruments
d) Counting Room Instruments

This equipment will be used, checked and calibrated by trained personnel according to in-plant procedures.

8.5.1 Portable Instruments

There will be sufficient types and quantities of portable instruments to provide adequate beta, gamma, and alpha surveys at LACBWR. This equipment will have the ability to detect these types of radiation over the potential ranges that will be present during SAFSTOR. Portable dose rate instruments will be source checked prior to use, and they will be calibrated semi-annually.

8.5.2 Installed Instrumentation

There will be sufficient types and quantities of installed instrumentation to provide continuous in-plant and effluent release monitoring. This will assure the safe reliable monitoring of both area dose rates and airborne activity concentration throughout the area. These instruments will be response tested monthly and calibrated once every 18 months.

8.5.3 Personnel Monitoring Instrumentation

Friskers and personnel instrumentation monitors will be provided throughout the plant to provide personnel contamination monitoring. These monitors will be of the type and sensitivities necessary to minimize the spread of in-plant contamination and prevent the introduction of contamination to outside areas. This equipment will be checked daily during normal workdays and calibrated semi-annually.

8.5.4 Counting Room Instrumentation

Laboratory equipment will be available to perform gross alpha and beta analyses and gamma isotopic analyses of samples collected in the plant. There will also be equipment available in a low background area to provide adequate analysis of environmental samples. A quality control program will be in effect for this equipment to ensure the accurate and proper operation of the equipment. This equipment will be traceable to NBS standards.

8.6 RADIOACTIVE WASTE HANDLING AND DISPOSAL

Radioactive waste at LACBWR during SAFSTOR will primarily consist of two different major types:

a) Resin
b) Dry active waste (DAW)

Waste generation will be maintained to as low as possible to minimize the volume generated for disposal.

8.6.1 Resin

Spent resin will be transferred to the spent resin receiving tank where it will be held until there is a sufficient quantity available for shipment to an approved burial site. The resin will then be transferred to an approved shipping container where it will be dewatered and made ready for shipment.

8.6.2 Dry Active Waste (DAW)

Any material used within the controlled area and disposed of will be considered radioactive and will be disposed of as DAW, unless it can be demonstrated to be within established releasable limits. The generation of this material will be maintained as low as possible to reduce the total waste volume generated onsite. The material generated will be compacted when possible to further reduce the volume and placed into approved shipping containers.

Disposal of all radioactive waste will be in accordance with all pertaining guidelines and will be made only to approved burial sites.

8.7 RECORDS

Records generated in the performance of the radiation protection program will be maintained as required to provide the necessary documentation of the program. These records will be maintained in a designated storage area.

8.8 INDUSTRIAL HEALTH AND SAFETY

LACBWR will continue to participate in Dairyland Power's industrial safety program as prescribed by the DPC Safety Department. These programs will include:

a) Accident prevention
b) Hazardous waste management and control
c) Asbestos control
d) Hearing conservation

9. SAFSTOR ACCIDENT ANALYSIS

9.1 INTRODUCTION

The probability of an accident occurring during the SAFSTOR period is considerably less than during plant operation. The focus of the potential accidents has also changed. During operation, the focus was on minimizing the plant transient and cooling the reactor core. During SAFSTOR, the only major concern is protecting the fuel in the Fuel Element Storage Well.

The fuel in the well, while not benign, is not as much a hazard as the fuel in the operating reactor was. Since April 30, 1987, the fission product inventory has decreased and the decay heat generation is significantly less. These factors reduce the consequences of any accident affecting the fuel. As time passes, the consequences will continue to decrease.

The reactor's design basis accidents were reviewed to determine which could still occur during SAFSTOR. Some other accident scenarios which were not previously considered design basis accidents were also evaluated. A list of 8 postulated accidents was identified. These events are:

- Spent Fuel Handling Accident
- Shipping Cask or Heavy Load Drop into FESW
- Loss of FESW Cooling
- FESW System Pipe Break
- Uncontrolled Liquid Waste Discharge
- Loss of Offsite Power
- Earthquakes
- Wind and Tornado

Each of these postulated events was evaluated based on the revised plant status to identify their potential consequences during the SAFSTOR period. The following sections discuss these accidents.

One additional event was examined - a fire. Fire protection is covered in Section 6.9. The potential safety consequences of any fire fall within the scope of other evaluated events.

9.2 SPENT FUEL HANDLING ACCIDENT

This accident postulates a fuel assembly falling from the hoist into the Fuel Element Storage Well. The probability of this accident is extremely small, since minimal fuel handling will be performed during the SAFSTOR period until the fuel assemblies are removed from the FESW. Periodic inspections may be conducted during the years the fuel remains onsite. In the almost 20 years of operation and associated fuel handling at LACBWR, no fuel assemblies were ever dropped.

In this event, it is assumed that the cladding of all the pins in 2 fuel assemblies ruptures. The fuel handling crew evacuates when the local area radiation monitor alarms. Containment Building ventilation would isolate on high activity, but for this analysis, no containment integrity is assumed.

The assumptions used in evaluating this event during SAFSTOR were similar to those used in the FESW reracking analyses.[1,2] The fuel inventory calculated for October 1987 was used. The only significant gaseous fission product available for release is Kr-85. The plenum or gap Kr-85 represents about 15% (215.7 Curies) of the total Kr-85 in the fuel assembly. However, for conservatism and commensurate with Reference 1, 30% of the total Kr-85 activity, or 431.4 Curies, is assumed to be released in this accident scenario.

No credit was taken for decontamination in the FESW water or for containment integrity, so all the activity was assumed to be released into the environment. Meteorologically stable conditions at the Exclusion Area Boundary (1109 ft, 338m) were assumed, with a release duration of 2 hours commensurate with 10 CFR 100 and Regulatory Guides 1.24 and 1.25.

A stack release would be the most probable, but a ground release is not impossible given certain conditions. Therefore, offsite doses were calculated for 3 cases. The first is at the worst receptor location for an elevated release, which is 500m E of the Containment Building. The next case is the dose due to a ground level release at the Exclusion Area Boundary. The maximum offsite dose at the new proposed Emergency Planning Zone boundary[3] for a ground level release is also calculated. Adverse meteorology is assumed for all cases.

<u>Elevated Release</u>

Average Kr-85 Release Rate

$$\frac{431.4 \text{ Curies}}{2 \text{ hrs.} \times 3600 \text{ sec/hr}} = 6.00 \text{ E-2 Ci/sec}$$

Worst Case $\frac{X}{Q}$ for 0-2 hours at 500m E = 2.3 E-4 sec/m³

Kr-85 average concentration at 500m E

$$6.00 \text{ E-2 Ci/sec} \times 2.3 \text{ E-4 sec/m}^3 = 1.38 \text{ E-5 Ci/m}^3$$

Immersion Dose Conversion at 500m E

Kr-85 Gamma Whole Body Dose Factor (Regulatory Guide 1.109)

$$1.61 \text{ E+1} \ \frac{\text{mRem/yr}}{\text{uCi/m}^3} \times 10^6 \ \frac{\text{uCi}}{\text{Ci}} \times 1.142 \text{ E-4} \ \frac{\text{yr}}{\text{hr}} = 1{,}839 \ \frac{\text{mRem/hr}}{\text{Ci/m}^3}$$

Whole Body Dose at 500m E

$$1839 \ \frac{\text{mRem/hr}}{\text{Ci/m}^3} \times 1.38 \text{ E-5 Ci/m}^3 \times 2 \text{ hr} = 0.05 \text{ mRem}$$

Kr-85 Beta/Gamma Skin Dose Factor (Regulatory Guide 1.109)

$$1.34\ E{+}3\ \frac{mRem/yr}{uCi/m^3} \times \frac{10^6\ uCi}{Ci} \times 1.142\ E{-}4\ \frac{yr}{hr} = 1.53\ E5\ \frac{mRem/hr}{Ci/m^3}$$

Skin Dose at 500m E

$$1.53\ E5\ \frac{mRem/hr}{Ci/m^3} \times 1.38\ E{-}5\ Ci/m^3 \times 2\ hr = 4.2\ mRem$$

Ground Level Release at TAB

Worst Case $\frac{X}{Q}$ for 2 hrs at 338m NE or 338m SSE, using Regulatory Guide 1.25

$$2.2\ E{-}3\ \frac{sec}{m^3}$$

Whole Body Dose at 338m

0.49 mRem

Skin Dose at 338m

40.4 mRem

Ground Level Release at Proposed Emergency Planning Zone Boundary

Worst Case $\frac{X}{Q}$ for 2 hrs at 100m E

$$1.02\ E{-}2\ \frac{sec}{m^3}$$

Whole Body Dose at 100m E

2.25 mRem

Skin Dose at 100m E

187 mRem

As can be seen, the estimated maximum whole body dose is more than a factor of 11,000 below the 10 CFR 100 dose limit of 25 Rem (25,000 mRem) to the whole body within a 2-hour period.

9.3 SHIPPING CASK OR HEAVY LOAD DROP INTO FESW

This accident postulates a shipping cask or other heavy load falling into the Fuel Element Storage Well. Reference 1 stated that extensive local rack deformation and fuel damage would occur during a cask drop accident, but with an additional plate (that was installed during the reracking) in place, a dropped cask would not damage the pool liner or floor sufficiently to adversely affect the leak-tight integrity of the storage well (i.e., would not cause excessive water leakage from the FESW).

For this accident, it is postulated that all 333 spent fuel assemblies located in the FESW are damaged. The cladding of all the fuel pins ruptures. The same assumptions used in the Spent Fuel Handling Accident (Section 9.2) are used here. A total of 35,760 Curies of Kr-85 is released within the 2-hour period. The doses calculated are as follows:

Elevated Release

Whole Body Dose at 500m E

4.2 mRem

Skin Dose at 500m E

350 mRem

Ground Level Release at EAB

Whole Body Dose at 338m

40.2 mRem

Skin Dose at 338m

3.34 Rem

Ground Level Release at Proposed Emergency Planning Zone Boundary

Whole Body Dose at 100m E

186 mRem

Skin Dose at 100m E

15.6 Rem

As can be seen, the estimated offsite doses for the cask drop accident are below the 10 CFR 100 limits. The postulated maximum whole body dose is more than a factor of 100 below the 10 CFR 100 limit of 25 Rem (25,000 mRem).

9.4 LOSS OF FESW COOLING

This accident postulates a loss of FESW cooling. The most likely causes of a loss of cooling are:

1) Both FESW pumps fail or FESW piping has to be isolated for maintenance;

2) The Component Cooling Water (CCW) System is out of service due to failure of both pumps or other reason. The CCW System removes heat from the FESW cooler.

3) The Low Pressure Service Water (LPSW) System is out of service due to failure of both pumps or other reason. The LPSW System removes heat from the CCW coolers.

If the third possibility is the cause, cooling to the CCW coolers can be restored by cross-connecting the High Pressure Service Water System to the coolers, in lieu of LPSW.

A calculation was performed to determine FESW heatup rate if active cooling were lost. The decay heat generation rate for January 1, 1988, was used. Pool boiling would commence approximately 5 days after the loss of cooling. The tops of the control rods stored in the fuel racks would become uncovered after 13.6 days. After 23.1 days, the top of the fuel would be exposed. If water is added to the FESW at any time during this period, these consequences would be delayed.

Substantial time is therefore available for restoration of FESW cooling. No immediate action is necessary during this postulated accident.

9.5 FESW PIPE BREAK

This accident postulates a break in the FESW system piping, other than in the pump discharge piping between the redundant check valves and the pool liner. A load analysis was performed on this approximately 20 feet of piping. It was concluded that all stresses are within ASME Code allowable. (Reference 1 calls this line the spent fuel pool drain line.) The series check valves were added during the 1980 FESW reracking.

If the postulated break occurs, the lowest the FESW could drain is approximately 679'. At this level all spent fuel will remain covered. The control rods which are currently stored in the fuel racks will be partially uncovered. The tops of the control rods are about elevation 686'.

The operator would be alerted to this accident by receipt of the FESW Level Lo/High alarm. Any makeup water added may run out the break, depending on the size of the break.

A calculation has been performed to determine the radiation levels due to the exposed control rods. In the vicinity of most of the FESW piping and isolation valves, the radiation dose would not be substantially increased due to the loss of water.

A repair team should be able to access the break location or piping isolation valves and either isolate the break or effect temporarily repairs. FESW level could then be restored to normal.

There would be no immediate urgency to restore the level. The partially uncovered control rods only create a local problem. No offsite release is associated with this event. Active FESW cooling would be lost during this accident, but as discussed in Section 9.4, considerable time is available to take action. Due to the lesser water volume to act as the heat sink and reduced fuel coverage, less time would be available to restore cooling during this accident scenario than in just a loss of FESW cooling event, but boiling would not commence for more than 1 day. As with the loss of FESW cooling event, if water is added to the FESW, any consequences of water heatup can be delayed or prevented. Water can be added from the Demineralized Water System or the Overhead Storage Tank.

9.6 UNCONTROLLED WASTE WATER DISCHARGE

This accident postulates that an operator starts pumping a Waste Water or Retention Tank to the river which is not sampled or for which the sample was incorrectly analyzed. If the contents of the tank are of normal activity, this event will not be detected until the lineup is being secured after pumping, if then.

If the liquid in the tank is of high activity, the waste water monitor will alarm and the Auto Flow Control Valve (54-22-002) automatically will close, terminating the discharge. The Turbine Condenser Cooling Water Monitor will also alarm, if the activity is high enough. If the automatic valve does not close, an operator will try to close it from the Control Room. If it cannot be closed, an operator will close a local valve or secure the pump to terminate the discharge.

After the discharge is terminated, a sample of the tank will be taken to analyze the uncontrolled release. Waste water is diluted by LACBWR Circulating Water and Low Pressure Service Water flow, in addition to circulating water from the adjacent coal-fired plant, prior to being discharged into the river.

9.7 LOSS OF OFFSITE POWER

This accident postulates a loss of offsite power. If both Emergency Diesel Generators and a High Pressure Service Water (HPSW) Diesel start, FESW cooling can be provided and adequate instrumentation is available to monitor FESW conditions from the Control Room. All that is needed is for an operator to cross-connect HPSW to the Component Cooling Water (CCW) coolers.

If an HPSW Diesel and 1B Emergency Diesel Generator start, FESW cooling can be provided. If 1A Emergency Diesel Generator (EDG) starts, but 1B does not, adequate cooling can be provided only if the essential buses are tied together.

If one or more EDG's start, but neither HPSW diesel starts, no ultimate heat sink for the FESW would be available. The consequences would be the same as in the Loss of FESW Cooling Event (Section 9.4).

If neither EDG can be started, neither FESW or CCW pump can run. The consequences again are the same as a Loss of FESW Cooling Event, with the additional complication that some instrumentation will be lost immediately and others after the station batteries are depleted. The operator would have to check the FESW locally periodically.

As discussed in Section 9.4, the FESW would not start boiling for about 5 days, the tops of the control rods would become uncovered in 13.6 days and the tops of the fuel would be exposed in 23.1 days. Therefore, no immediate action needs to be taken and sufficient time is available to take corrective actions to restore power.

9.8 SEISMIC EVENT

This accident postulates that a design basis earthquake occurs. The magnitude of the seismic event and damage incurred is the same as that assumed during the Systematic Evaluation Program (SEP) and the Consequence Study prepared as part of the SEP Integrated Assessment (References 4-7). The major concern of the previous evaluation was to safely shut down the plant and maintain adequate core cooling to prevent fuel damage. The focus now is to prevent damage to the fuel stored in the Fuel Element Storage Well.

Seismic analysis has shown the Containment Building structure, LACBWR stack and Genoa Unit 3 stack are capable of withstanding the worst postulated seismic event at the LACBWR site. Reference 1 documented that the storage well, itself, the racks and the bottom-entry line between the check valves and the storage well can withstand the postulated loads.

The potential consequences of most interest due to a seismic event could include loss of all offsite and onsite power and a break in the FESW System piping. This event, therefore, can be considered as a combination of a Loss of Power Event (Section 9.7) and FESW Line Break (Section 9.5). As with these individual events, considerable time is available for response to a seismic event, with the FESW System pipe break requiring the earlier response. Access to the break location may be more difficult following a seismic event due to failure of other equipment in the plant. The time available, though, should be more than sufficient to initiate mitigating actions. (Refer to Section 9.5).

9.9 WIND AND TORNADO

This accident postulates that design basis high wind or tornado event occurs. The magnitude of the event and damage incurred is the same as that assumed during the Systematic Evaluation Program (SEP) and the Consequence Study prepared as part of the SEP Integrated Assessment (References 4-9). The major concern of the previous analyses was to ensure that adequate cooling of the reactor core was maintained. The focus now is to prevent damage to the fuel stored in the Fuel Element Storage Well.

The previous evaluations determined that the Containment Building would withstand this event. The Turbine Building, Diesel Building, Cribhouse and Switchyard may be damaged. The probability of the LACBWR or Genoa Unit 3 stacks failing and impacting the Containment Building was determined to be low enough that it need not be considered. Personnel outside the Containment Building may not survive.

The potential plant consequence of primary concern is the loss of all offsite and onsite power. As discussed in Section 9.7, Loss of Offsite Power, considerable time is available before action must be taken to protect the fuel.

9.10 REFERENCES

1) NRC Letter, Ziemann to Linder, dated February 4, 1980.

2) NRC Letter, Reid to Madgett, dated October 22, 1975.

3) DPC Letter, Taylor to Document Control Desk, LAC-12377, dated September 29, 1987.

4) DPC Letter, Linder to Paulson, LAC-10251, dated October 11, 1984.

5) NRC Letter, Zwolinski to Linder, dated January 16, 1985.

6) DPC Letter, Linder to Zwolinski, LAC-10639, dated March 15, 1985.

7) NRC Letter, Zwolinski to Taylor, dated September 9, 1986.

8) DPC Letter, Taylor to Zwolinski, LAC-12052, dated January 14, 1987.

9) NRC Letter, Bernero to Taylor, dated April 6, 1987.

10. SAFSTOR OPERATOR TRAINING AND CERTIFICATION PROGRAM

10.1 INTRODUCTION

This program describes the training and certification for supervisors and operators associated with the maintenance and monitoring of the La Crosse Boiling Water Reactor (LACBWR) in the SAFSTOR mode consistent with its possession-only license.

10.2 APPLICABILITY

The LACBWR Technical Specifications will require that certain operations associated with the maintenance and handling of reactor spent fuel be performed by or under the supervision of persons certified by the Plant Superintendent or his delegate. The following members of the plant staff (as a minimum) shall be certified in accordance with this program:

- Plant Superintendent
- Operations Supervisor
- Shift Supervisors
- Selected operators who shall be performing duties requiring certified operators.

10.3 INITIAL CERTIFICATION

Certification candidates shall participate in a training program covering the following topic areas:

a) Reactor Theory (as applicable to the storage and handling of spent reactor fuel)

b) Spent Fuel Handling and Storage Equipment - Design and Operating Characteristics

c) Monitoring and Control Systems

d) Radiation Protection

e) Normal and Emergency Procedures

f) Administrative Controls applicable during the SAFSTOR period

Reactor Theory training will include characteristics of the stored spent fuel, subcritical multiplication, factors affecting reactivity and criticality, and the basis for fuel handing restrictions and procedures.

The design and operating characteristics will include training in the functions and use of fuel handling tools, cranes, the fuel element storage well, and pool service systems and equipment. Prior to shipments of spent fuel this training will include shipping casks, cask handing equipment, and procedures.

Monitoring and Control Systems will include training on the Fuel Element Storage Well monitoring systems and area radiation monitors.

Radiation protection training will include theory of radioactive emissions, control of radiation exposure, use of radiation detection and monitoring equipment, protective clothing and respiratory protection, and contamination control procedures. Training will emphasize the principles and practices associated with maintaining exposures as low as reasonably achievable (ALARA).

Normal and Emergency Procedure Training will include the Emergency Plan and any operations and emergency procedures associated with the operation of LACBWR systems and equipment during SAFSTOR. This area shall also include training in the handling and processing of radioactive wastes.

Administrative Control Training will include LACBWR Technical Specifications, Security Plan, Quality Assurance Program Description and plant administrative procedures associated with the operation, surveillance, and maintenance of LACBWR.

Training will be provided through a combination of classroom instruction, audio-visual instruction, self-study, and on-the-job training.

Satisfactory completion of the training shall be based on passing of a comprehensive written examination including each of the above areas and an oral examination. Minimum passing grade for the written examination shall be 70% in each area and 80% overall. The oral examination shall be administered by a member of the plant management staff. Results of the oral examination shall be on a pass/fail basis. Weaknesses noted as a result of the written or oral examination shall be documented and remedial training provided.

10.4 PROFICIENCY TRAINING AND TESTING

Proficiency training shall be used to maintain the qualification level of certified personnel. Proficiency training will include periodic training through the use of classroom training, audio/visual instruction, self-study assignments, and/or on-the-job training. Frequency and topics to be included in the proficiency training will depend on actual activities planned or in progress and identified weaknesses. As a minimum, training in the six areas included in the initial certification program shall be covered at least once every 2 years.

A biennial written examination and an annual oral examination shall be used to demonstrate the proficiency of certified personnel. Examinations will be similar to, but not as comprehensive as, the initial certification examinations. Minimum passing grade for proficiency examinations shall be 70% in each section and 80% overall. Oral examinations shall be on a pass/fail basis.

10.5 CERTIFICATION

Upon successful completion of the initial certification training program, the Plant Superintendent or his delegate shall certify the individual as a Certified Fuel Handler. Normally an employee will complete the initial certification within one year after entering the program. After initial certification, personnel will be recertified every 2 years based on the successful completion of the Proficiency Training and Testing Program.

10.6 PHYSICAL REQUIREMENTS

As a prerequisite to acceptance into the training program and for recertification, a candidate must successfully pass a medical examination designed to ensure that the candidate is in generally good health and is otherwise physically qualified to safely perform the assigned work. Minor correctable health deficiencies, such as eyesight or hearing, will not per se prevent certification.

The medical examination will meet or exceed the requirements of ANSI Standard N546-1976, "American National Standard - Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants."

10.7 DOCUMENTATION

Initial Certification and Proficiency Training shall be documented and maintained for certified personnel while employed at LACBWR. The records shall include the dates of training, results of all quizzes and examinations, copies of written examinations, oral examination records, and information on results of physical examinations.

PRELIMINARY

DECON PLAN

FOR THE

LA CROSSE BOILING WATER REACTOR
(LACBWR)

December 1987

INTRODUCTION

The purpose of this Preliminary DECON Plan for the La Crosse Boiling Water Reactor (LACBWR) is to state Dairyland Power Cooperative's (DPC) intentions regarding the ultimate decommissioning of LACBWR. DPC is planning on a 30-50 year SAFSTOR period to be followed by a DECON effort to decontaminate and dismantle remaining contaminated systems and structures as necessary to achieve a radiologically releasable site and termination of the Nuclear Regulatory Commission (NRC) license.

A detailed Decommissioning Plan is being submitted to describe the status of LACBWR during the SAFSTOR period. The Decommissioning Plan covers the current plant status and activities and programs during the SAFSTOR period. An estimated radiological characterization is provided.

This Preliminary DECON Plan is strictly an overview of the final decommissioning period. It would be premature at this time to detail decontamination methods or the specific organization to be used for the DECON staff. Prior to commencement of the final DECON effort, a detailed DECON Plan will be prepared and submitted to the NRC.

PLANNING

DPC is currently planning for a 30-50 year SAFSTOR period, followed by a final DECON effort. The DECON stage cannot commence until after the irradiated fuel is removed from LACBWR. The fuel is expected to be stored onsite for approximately 20 years, until a federal or other suitable repository is available.

Once the fuel is removed from site, a revised SAFSTOR Decommissioning Plan will be prepared. At that time the duration of the SAFSTOR period will be re-evaluated. Prior to the end of the SAFSTOR period, detailed planning will be conducted for the DECON effort.

A detailed radiological characterization will be conducted during the DECON planning. The characterization will include an extensive plant radiation survey and sampling of remaining plant systems, structures, site soil, the plant outfall, and other areas associated with plant operation. During the SAFSTOR period, routine plant radiation surveys and environmental monitoring will be conducted. This information will also be used for the radiological characterization.

Various methodologies will be evaluated for the decontamination and dismantling activities. The state-of-the-art methodologies at the time of the LACBWR DECON may be considerably different than those available today. Therefore, the processes to be used will be selected at that time. It is expected that several methodologies will be utilized dependent on the specifics of the item being decontaminated or dismantled.

Following completion of the radiological characterization and planning for the DECON effort, a detailed DECON Plan will be submitted to the NRC.

ORGANIZATION

An organization will be established to conduct the DECON effort. The responsibility for the project will remain with Dairyland Power Cooperative, though a part of the work and project management may be performed by outside contractors.

Decommissioning activities will continue to be performed in accordance with a Quality Assurance Program during the DECON effort. The Quality Assurance Program may be revised to make it more compatible with the activities which will be conducted during the DECON project.

SAFETY

A radiation protection program and an industrial health and safety program will be implemented during the DECON period. It is expected that these programs will be continuations of the ones conducted during SAFSTOR. The programs will be modified as necessary to cover the increased scope of work during the DECON project.

All major DECON activities will be performed as controlled evolutions. The safety of the public and of the workers will be considered in the planning stage for the DECON effort.

The basic goal of the radiation protection program will still be to ensure the health and safety of LACBWR workers and of the public. The philosophy of maintaining dose to the workers As Low As Reasonably Achievable (ALARA) will continue to be followed. The program will continue to encompass actions to both control and monitor to workers radiation exposure.

The basic goal of the industrial health and safety program will also still be to ensure the health and safety of LACBWR workers. The program will continue to encompass actions to both control and monitor the hazards to which workers are exposed.

RADIOACTIVE WASTE MANAGEMENT

The radioactive waste management program during the DECON project will be a continuation of that used during the SAFSTOR period. It is expected that the waste handling systems will be among the last to be dismantled and temporary facilities will be established as necessary.

Contaminated waste generation will be maintained as low as possible to minimize the volume requiring disposition as radioactive waste. Disposal of all waste will be in accordance with applicable regulations.

ENVIRONMENTAL IMPACT

A review will be conducted of the environmental impact of the planned DECON activities. A supplement to the Environmental Report for the La Crosse Boiling Water Reactor will be submitted addressing any significant environmental change associated with the planned DECON activities.

COST

Every 5 years during the SAFSTOR period, a review of the decommissioning cost estimate will be performed in order to assure sufficient funds are available for the final DECON. The detailed DECON Plan will address the final cost estimate and availability of funds.

SITE RELEASE

Following completion of the planned decontamination and dismantling activities, a final radiological survey of the site and associated areas will be conducted to verify that all residual LACBWR generated radioactivity levels are within established limits. The detailed radiological characterization will be used to help determine the selection of survey and sample locations. Specific attention will be paid to areas at which any spills occurred during plant life. If any areas are identified to be above the established limits, additional actions will be taken to reduce the residual radioactivity.

The disposition of any structures remaining after the residual radioactivity levels are within established limits will depend on what activities DPC chooses to perform at that time. The remaining facilities may be used for other Cooperative purposes or selectively demolished.

A final report will be submitted to the NRC following completion of a final satisfactory radiological survey.