WILLIAM L. BERG
President and CEO

DAIRYLAND POWER
COOPERATIVE

November 29, 2005

In reply, please refer to LAC-13887

DOCKET NO. 50-409

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

SUBJECT: Dairyland Power Cooperative
La Crosse Boiling Water Reactor (LACBWR)
Possession-Only License DPR-45
Annual Decommissioning Plan Revision

REFERENCES:
(1) DPC Letter, Taylor to Document Control Desk, LAC-12460, dated December 21, 1987 (original submittal of LACBWR's Decommissioning Plan)
(2) NRC Letter, Erickson to Berg, dated August 7, 1991, issuing Order to Authorize Decommissioning of LACBWR
(3) NRC Letter, Brown to Berg, dated September 15, 1994, modifying Decommissioning Order

The annual update of the LACBWR Decommissioning Plan has been completed, and the pages with changes and their explanations are included with this letter. Each page with a change will have a bar in the right-hand margin to designate the location of the change. None of the changes was determined to require prior NRC approval, and they have been reviewed by both the plant Operations Review Committee and the independent Safety Review Committee.

The individual pages requiring revision are enclosed with this letter. Please substitute these revised pages in your copy(ies) of the LACBWR Decommissioning Plan. Reasons for the changes are listed on a separate enclosure.

If you have any questions concerning any of these changes, please contact Jeff Mc Rill of my staff at 608-689-4202.

Sincerely,

DAIRYLAND POWER COOPERATIVE

William L Berg

William L. Berg, President & CEO

WLB:JBM:dh
Enclosures

cc: Kris Banovac
NRC Project Mgr.

NMSS01

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2005 LACBWR Decommissioning Plan Review

Cover Page	Update revision date.
Page 3-8 Through Page 3-16	Format of pages is updated and pages are reissued with no other changes.
Section 3 Figures	Figures 3.1, 3.2, 3.3, 3.6, and 3.7 have page format updated and are reissued with no other changes.
Section 4 Figures	Figures 4.1, 4.2, 4.3, and 4.4 have page format updated and are reissued with no other changes.
Page 5-1 And Page 5-2	Format of pages is updated and pages are reissued with no other changes.
Page 5-3	<u>Section 5.2.1, Reactor Vessel and Internals</u>: The last two sentences of "System Status" are deleted saying, *"The reactor vessel is capable of being refilled. The control rods may be removed to the Fuel Element Storage Well or a licensed facility during SAFSTOR."* The following content is added as new paragraphs to the end of "System Status" to provide information of the LACBWR Reactor Pressure Vessel Removal Project: ***"The LACBWR Reactor Pressure Vessel Removal Project was begun in August 2005. The reactor vessel with head installed, internals intact, and 29 control rods in place will be filled with low density cellular concrete. Attachments to the reactor vessel flange will be removed to a diameter of 119 inches. All other nozzles and appurtenances will be cut to within the diameter of the flange. Under-vessel nozzles and appurtenances will be removed from an envelope of within 6 inches of bottom dead center of the reactor vessel shell bottom.*** ***The reactor pressure vessel will be removed from the Reactor Building and packaged for shipment by rail to the Barnwell Waste Management Facility in South Carolina for disposal by June 2007."***
Page 5-4	<u>Section 5.2.2, Forced Circulation System</u>: The following content is added as a new paragraph to the end of "System Status" to provide information of the LACBWR Reactor Pressure Vessel Removal Project: ***"All 16-inch and 20-inch forced circulation system piping will be filled with low density cellular concrete as part of the Reactor Pressure Vessel Removal Project begun in August 2005. Four 16-inch forced circulation inlet nozzles and four 16-inch outlet nozzles will be cut to allow removal of the reactor pressure vessel."***

Page 5-7 Section 5.2.5, Emergency Core Spray System: The last sentences "System Status" is deleted saying, *"The low pressure supply is retained to provide Reactor Vessel refill capability."* This low pressure supply line will be removed as interference to the removal of the reactor pressure vessel.

Page 5-10 Section 5.2.8, Alternate Core Spray System: "System Status" is revised by the following, *"The Alternate Core Spray System is not required to be operational in SAFSTOR. The manual isolation valve to the Reactor Building is closed.* ***The 6-inch supply line to the reactor pressure vessel head will be removed as interference to removal of the reactor pressure vessel. Motor operated valves and instrumentation in the Turbine Building have been electrically removed. System components continue to serve requirements of the HPSW System. These components will be designated as part of the HPSW System in the near future."*** Purpose of change is to provide information of the LACBWR Reactor Pressure Vessel Removal Project.

Page 5-13 Section 5.2.11, Fuel Element Storage Well System: "System Status" is revised to state, *"The Fuel Element Storage Well contains 333 irradiated fuel elements and will remain in operation as part of the SAFSTOR Program as long as wet fuel storage or wet fuel handling is necessary.* ***Also stored in the well are 10 control rods, 2 antimony-beryllium startup sources, 24 stainless steel fuel element shroud cans, and 73 zirconium alloy fuel element shroud cans. These components will be removed, packaged, and disposed of. Removal of irradiated hardware and other B&C wastes has been included in the scope of work during the Reactor Pressure Vessel Removal Project."*** Purpose of change is to provide information of the LACBWR Reactor Pressure Vessel Removal Project.

Page 5-23 Section 5.2.21, High Pressure Service Water System: The fourth sentence is revised by stating, *"With the motor-driven pump cycling in automatic, HPSW system pressure is maintained* ***110 to 135 psig at the expansion tank pressure switch elevation, 25 feet above site grade elevation."*** The sixth sentence is revised by stating, *"Backup supply is available from two HPSW diesel pumps.* ***1A HPSW Diesel Pump will start automatically if system pressure decreases to 90 psig. 1B HPSW Diesel Pump will start automatically if system pressure decreases to 80 psig."*** In the second paragraph, the following is added to the end, ***"The external loop is also cross-connected with the Fire Suppression System of the adjacent coal-fired generating facility, Genoa Unit 3. This cross-connect provides excess HPSW diesel pump capacity to this operating plant."*** These changes are due to work completed under an approved facility change.

Page 5-24 Section 5.2.22, Circulating Water System: Second paragraph the phrase, *"Genoa No. 3 Plant"* is changed to read *"Genoa Unit 3."* Change is for consistency in terminology.

Page 5-37	Section 5.2.33, Electrical Power Distribution: "System Status," second paragraph is revised to state, *"The Electrical Power Distribution System will be modified significantly. These modifications will configure the system to provide backup diesel generator power to the entire 480-V AC system. Facility power use has decreased below the capacity of the diesel generators making changes possible. Planned changes will ensure continued operation of normal lighting systems, air conditioning systems, and sanitary well water supply for habitability reasons and plumbing system use. Further simplification and reliability will be gained."* Purpose of change is to provide information of upcoming electrical configuration change. This change will move breaker interlocks to allow entire 480-V AC system to be energized automatically by diesel generator power. Subsequent Decommissioning Plan revisions will reflect changes and simplification of the Electrical Power Distribution System.
Page 6-2 Through Page 6-6	Format of page 6-2 is updated and pages following are reissued due to content shift with no other changes.
Page 6-9	Page is reissued to include some content of page following with no other changes.
Page 6-10	Section 6.5, Quality Assurance: A new fourth paragraph is added to provide information contained in the QAPD applicable to all SAFSTOR activities at LACBWR. The new paragraph states, *"Scheduled activities during SAFSTOR shall be performed within schedule intervals. A schedule interval is a time frame within which each scheduled activity shall be performed, with a maximum allowable extension not to exceed 25 percent of the schedule interval."* Section 6.6, Schedule: A new third paragraph is added to provide information of the LACBWR Reactor Pressure Vessel Removal Project: *"Section 7.3.4 describes a major project undertaken at LACBWR. Duratek proposed to DPC in April 2005 that disposal of the Reactor Pressure Vessel (RPV) could proceed with fuel in the Reactor Building spent fuel pool. This disposal could occur prior to the Barnwell Waste Management Facility (BWMF) closing to out of compact waste in July 2008. In April 2005, DPC commissioned Duratek to study the feasibility of disposing of the RPV, intact, with existing internals at the BWMF. In August 2005, the results of this study led DPC to the decision to go forward with the actual removal of the RPV."*
Page 6-11	Section 6.6, Schedule: Continuing section from previous page; the second paragraph of page 6-11, after the first sentence, a new second sentence is added and the final sentence revised to state, *"The Nuclear Regulatory Commission authorized the NRC staff to issue PFS a license on September 9, 2005. LACBWR spent fuel removal strategy and cask storage technology are being*

evaluated." Purpose of change is to update status of Private Fuel Storage ISFSI licensing and LACBWR spent fuel removal.

Page 6-12 Through Page 6-15	Pages are reissued due to content shift from previous page with no other changes.
Page 6-16	Section 6.9.2.4, Fire Suppression Water System: In the first paragraph, third sentence, *"<60 PSIG"* is revised to state, ***"<90 PSIG for HPSW Diesel Pump 1A or <80 PSIG for HPSW Diesel Pump 1B."*** In the same sentence, the phrase, *"Genoa Station No. 3 (G-3)"* is changed to read *"Genoa Unit 3."* These changes are due to work completed under an approved facility change and for consistency.
Page 6-17 Through Page 6-19	Pages are reissued due to content shift from previous page with no other changes.
Page 7-4	Section 7.3.4, Research: Section is deleted because no Aging Research Program has been performed since shutdown in 1987 and no plans are apparent for completing any such research.

New content is added at Section 7.3.4 to describe reactor pressure vessel removal. Section 7.3.4 is renamed and contains the following:

Section 7.3.4, Reactor Pressure Vessel Removal

"Based on a feasibility study, DPC has entered into contract agreement with Duratek, Inc. for the removal and subsequent disposal of the intact Reactor Pressure Vessel (RPV) at the Barnwell Waste Management Facility in South Carolina. The RPV disposal is expected to be completed in the spring of 2007. DPC has included in the scope of work during this project removal of irradiated hardware and other B&C wastes.

The Decommissioning Plan discusses and provides for removal of unused equipment and plant system components, in accordance with 10 CFR 50.59, during SAFSTOR. The RPV removal is not specifically addressed in the decommissioning plan schedule. The removal of this large component, as defined in 10 CFR 50.2, is an activity requiring notice be made pursuant to 10 CFR 50.82, Termination of License, (a)(7). This notice was made by submittal to the NRC, in writing August 18, 2005. A copy of the submittal was sent to the affected State(s) before performing any decommissioning activity inconsistent with, or making any significant schedule change from, those actions and schedules described in the PSDAR, including changes that significantly increase the decommissioning cost."

Page 7-5 And Page 7-6	Pages are reissued due to content shift from previous page with no other changes.
Page 9-2 Through Page 9-3	Section 9.2, Spent Fuel Handling Accident: The curie content remaining as of *October 2004* and calculated values for Whole Body Dose and Skin Dose as of *October 2004* are updated to ***October 2005***.
Page 9-4	Section 9.3, Shipping Cask or Heavy Load Drop into FESW: The curie content remaining as of *October 2004* and calculated values for Whole Body Dose and Skin Dose as of *October 2004* are updated to ***October 2005.***
Page 10-3	Format of page is updated and page is reissued with no other changes.

INITIAL SITE CHARACTERIZATION SURVEY FOR SAFSTOR (LAC-TR-138):

Cover Page	Update revision date.
Page 24 Through Page 28	Curie content values stated in pages 24-28 are updated. These pages of Attachments 1, 2, 3 have been decay-corrected to ***October 2005,*** replacing pages that had been decay-corrected to *October 2004*.

LA CROSSE BOILING WATER REACTOR (LACBWR)

NRC Docket No. 50-409

DAIRYLAND POWER
COOPERATIVE

TO: NRC Washington CONTROLLED DISTRIBUTION NO. 53

FROM: LACBWR Plant Manager 12/8/2005

SUBJECT: Changes to LACBWR Controlling Documents

I. The following documents have been revised:

DECOMMISSIONING PLAN, revised November 2005

Remove and replace the following pages:
Title Page
pages 3-8 thru 3-16,
Figures 3.1 thru 3.3, Figures 3.6 & 3.7
Figures 4.1 thru 4.4
pages 5-1 thru 5-4, 5-7, 5-10, 5-13, 5-23, 5-24, 5-37
pages 6-2 thru 6-6, 6-9 thru 6-19

Remove pages 7-4 & 7-5, replace with pages 7-4 thru 7-6
Remove and replace pages 9-2 thru 9-4, and page 10-3

SITE CHARACTERIZATION SURVEY

Remove and replace the following pages:
Title page
pages 24 thru 28

☒ The material listed above is transmitted herewith. Please verify receipt of all listed material, destroy superseded material, and sign below to acknowledge receipt.

☐ The material listed above has been placed in your binder.

☐ Please review listed material, notify your personnel of changes, and sign below to acknowledge your review and notification of personnel. [To be checked for supervisors for department specific procedures and LACBWR Technical Specifications.]

☐ The material listed above has been changed. [To be checked for supervisors when materials applicable to other departments are issued to them.]

/S/______________________ DATE______________

Please return this notification to the LACBWR Secretary within ten (10) working days.

A Touchstone Energy® Cooperative

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LA CROSSE BOILING WATER REACTOR

(LACBWR)

DECOMMISSIONING PLAN

Revised
November 2005

DAIRYLAND POWER COOPERATIVE
LA CROSSE BOILING WATER REACTOR (LACBWR)
4601 State Road 35
Genoa, WI 54632-8846

LA CROSSE BOILING WATER REACTOR

(LACBWR)

DECOMMISSIONING

PLAN

Revised
November 2005

DAIRYLAND POWER COOPERATIVE
LA CROSSE BOILING WATER REACTOR (LACBWR)
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Genoa, WI 54632-8846

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 Palezoic 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 La Fayette 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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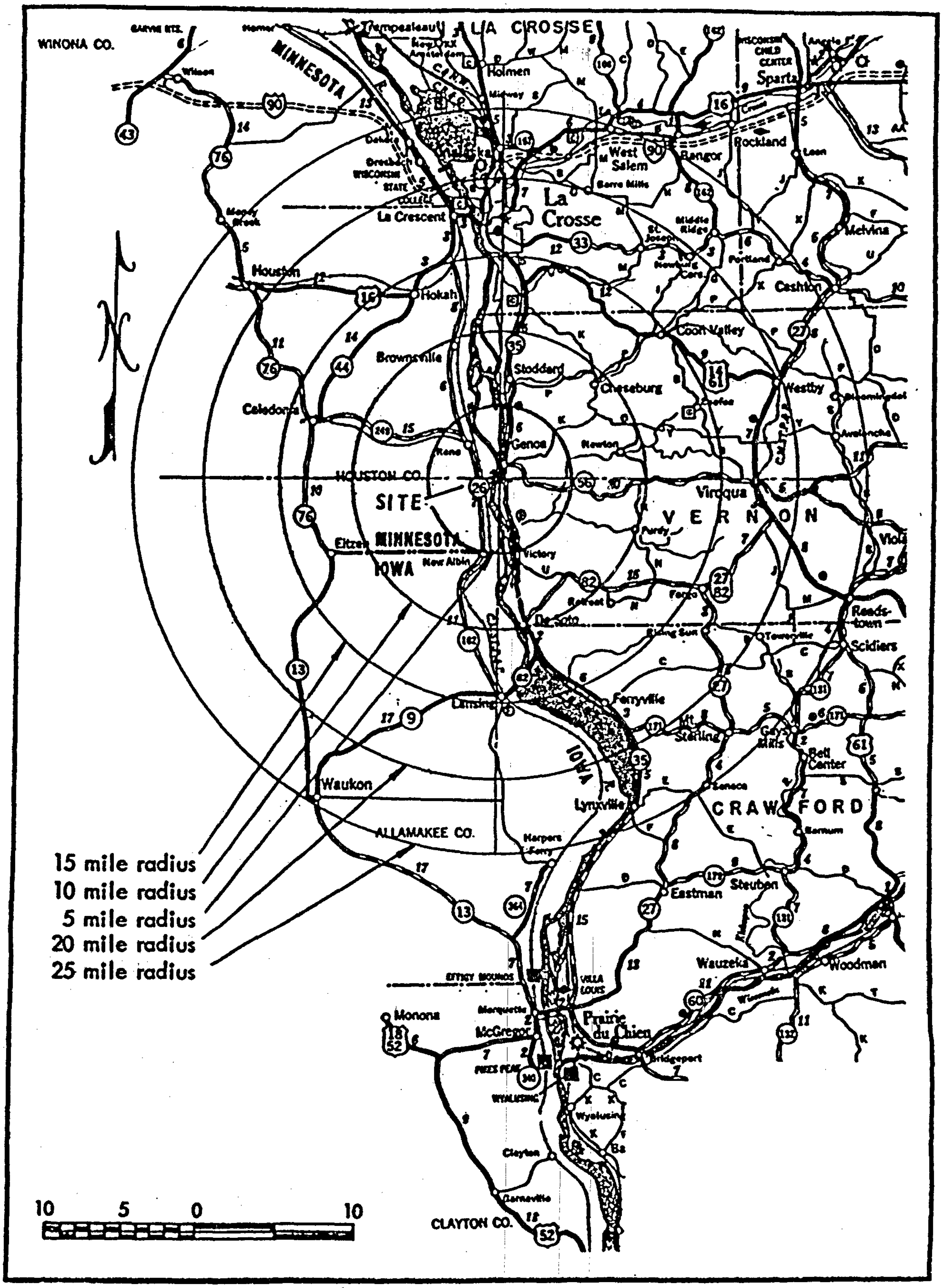
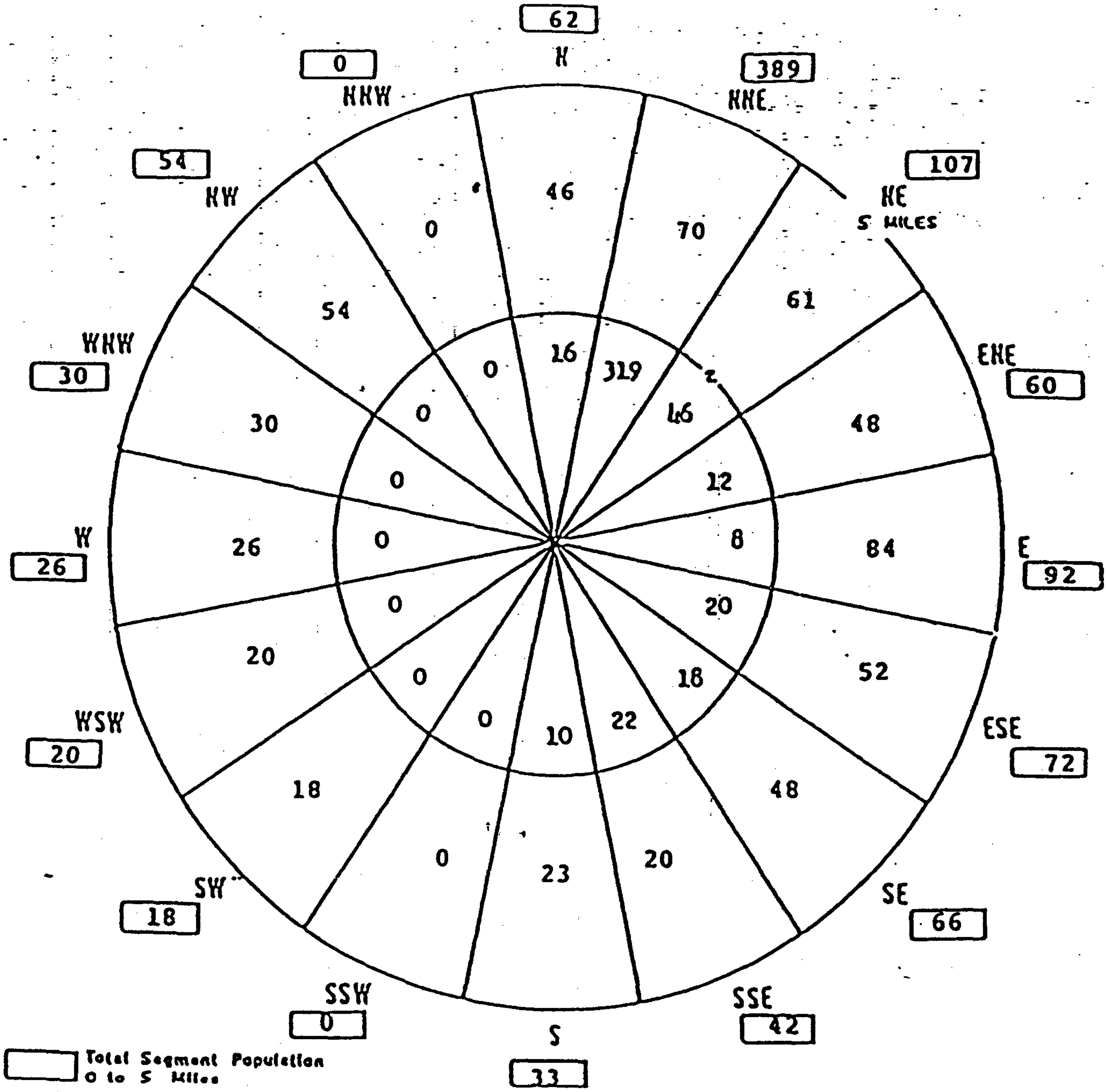


<u>General Site Location Map</u>

FIGURE 3.1

D-PLAN



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)

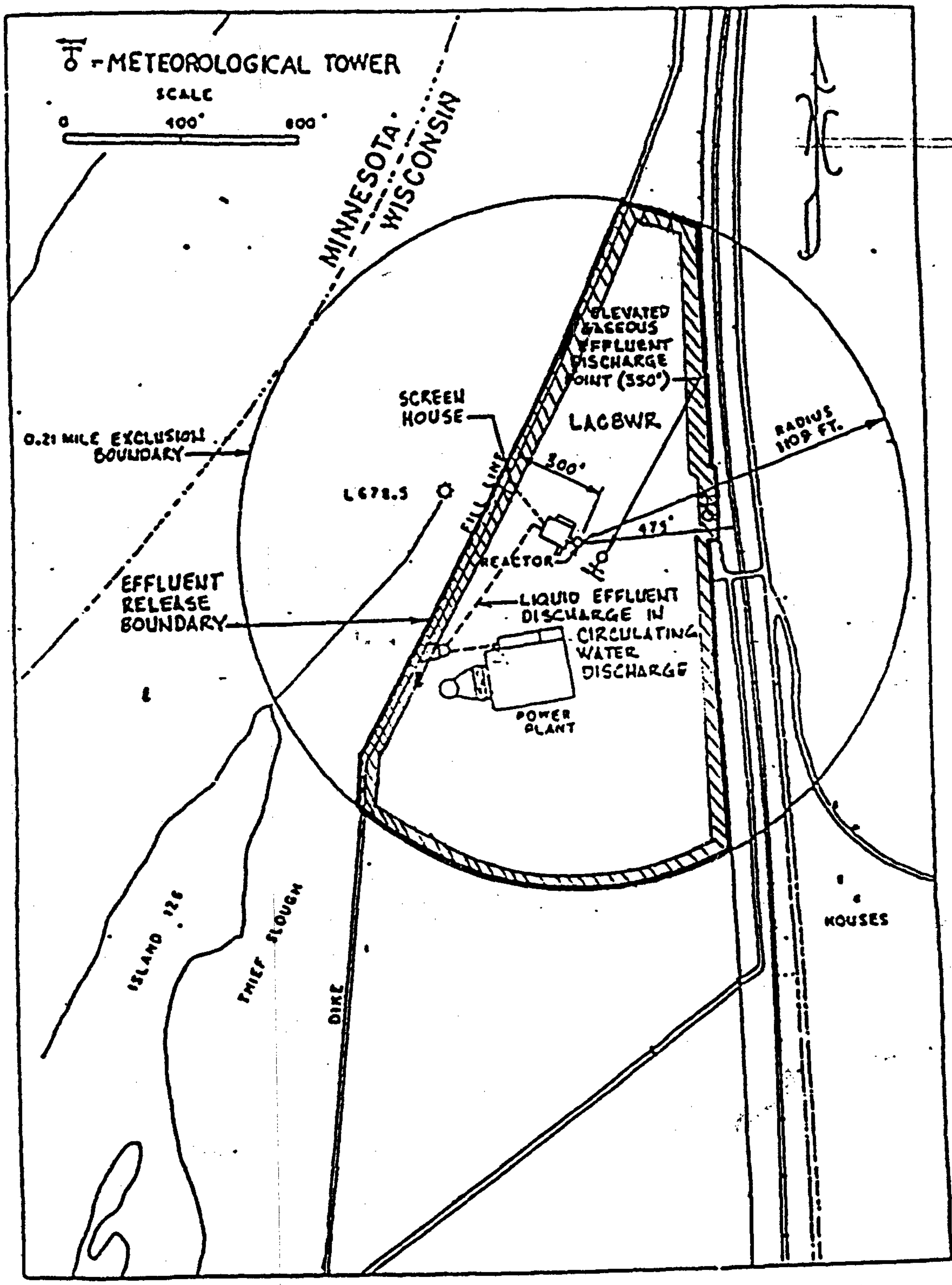
MN = 148 (13.82%) WI = 923 (86.18%)

IA = 0

Population Dispersion

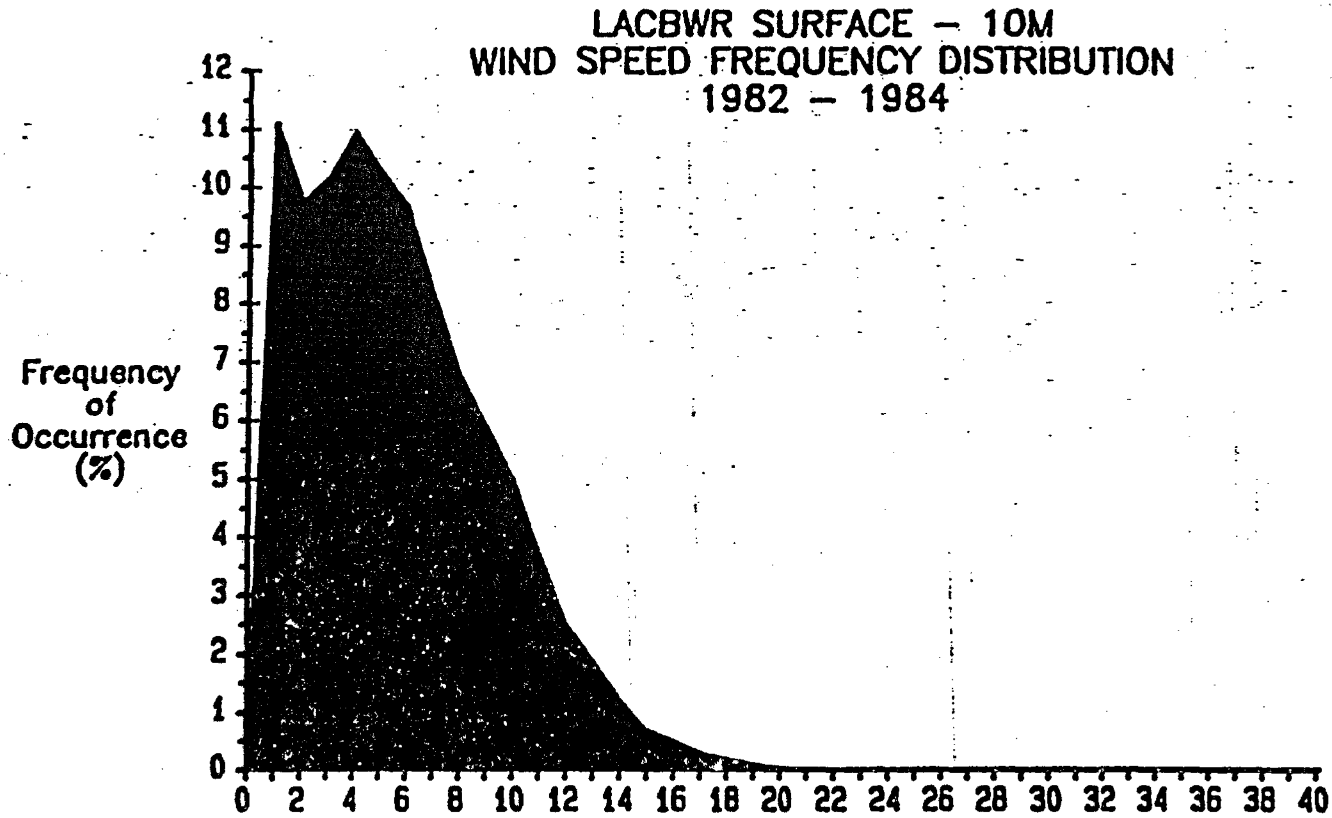
FIGURE 3.2

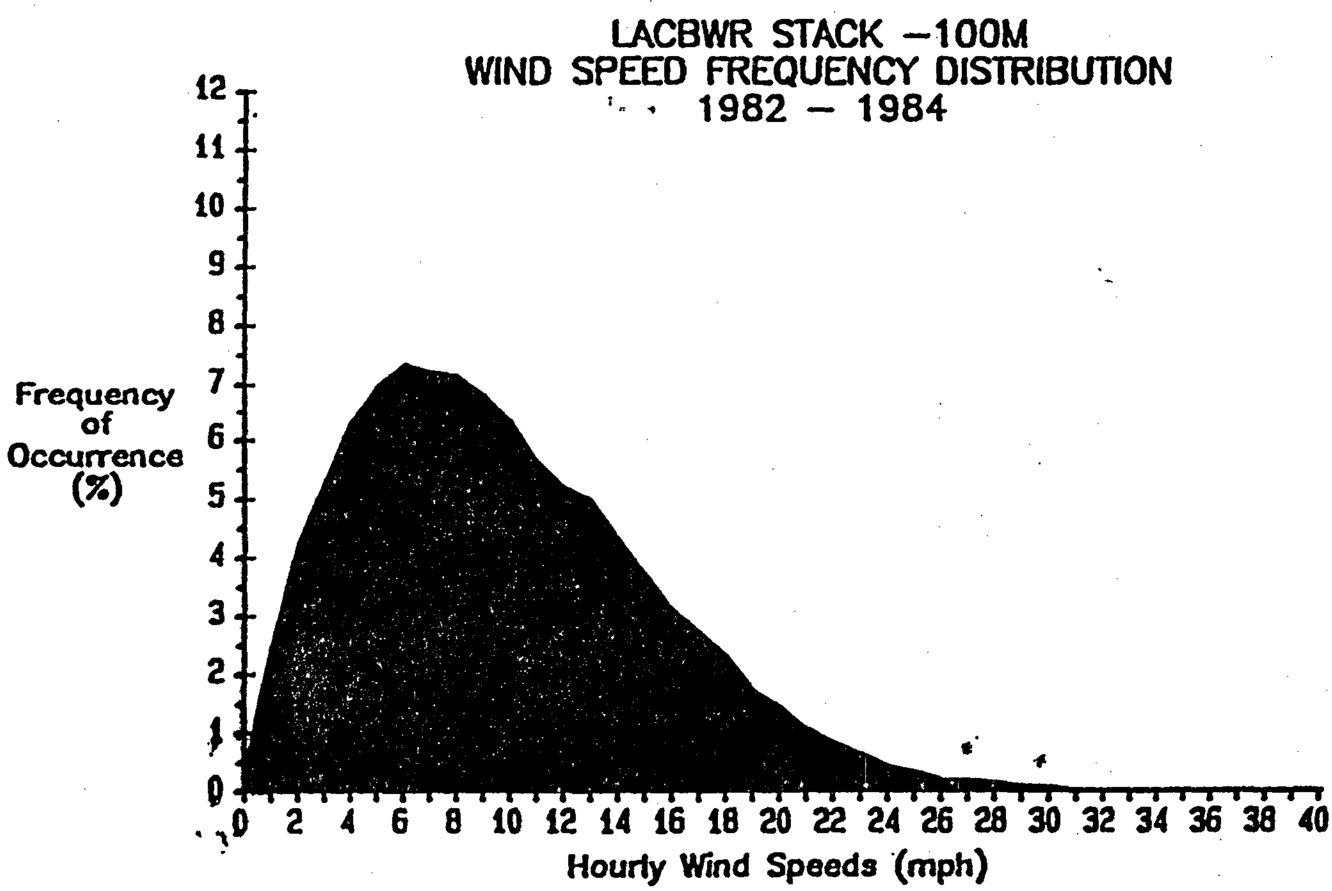
D-PLAN



Effluent Release Boundary

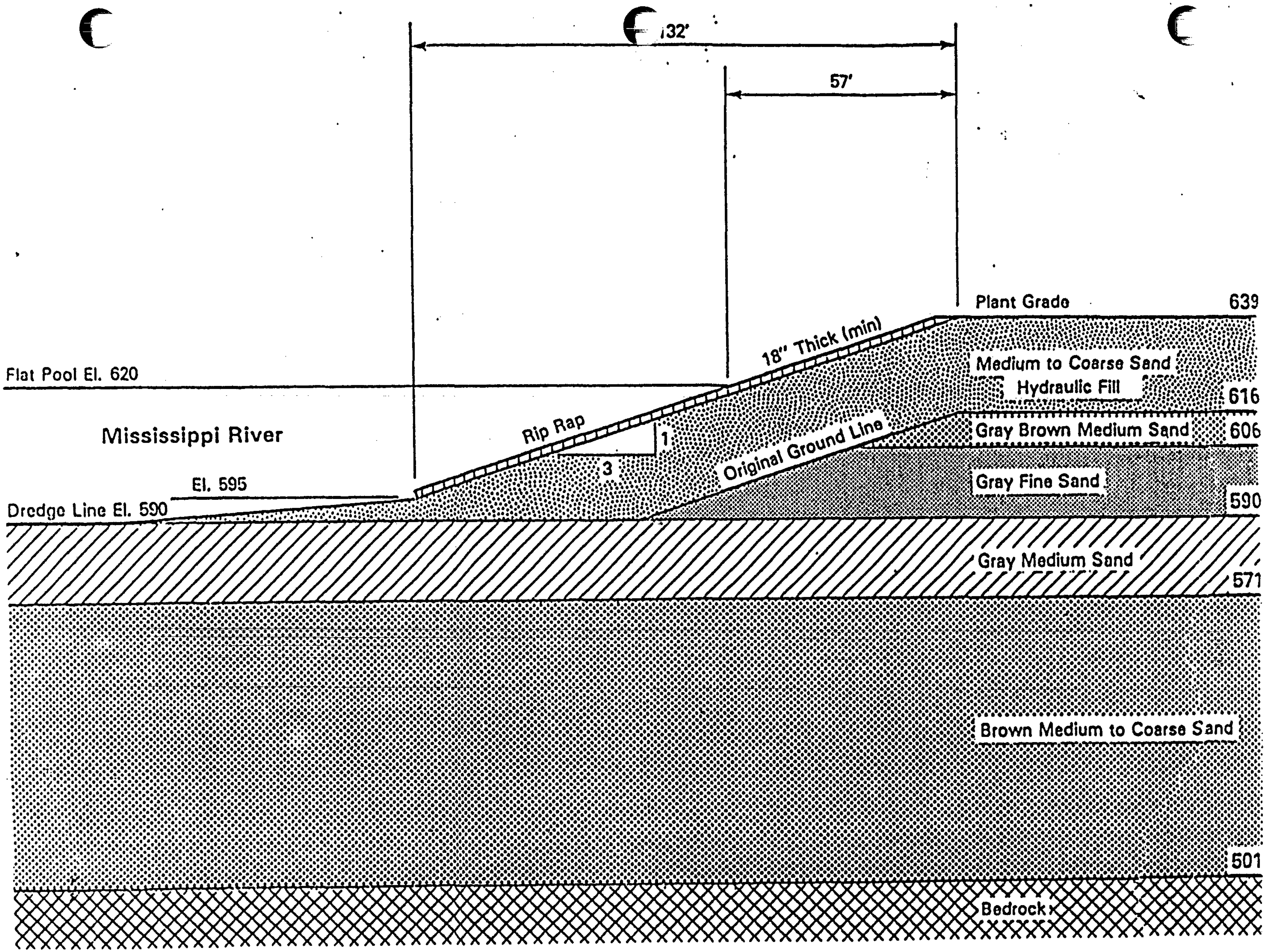
FIGURE 3.3





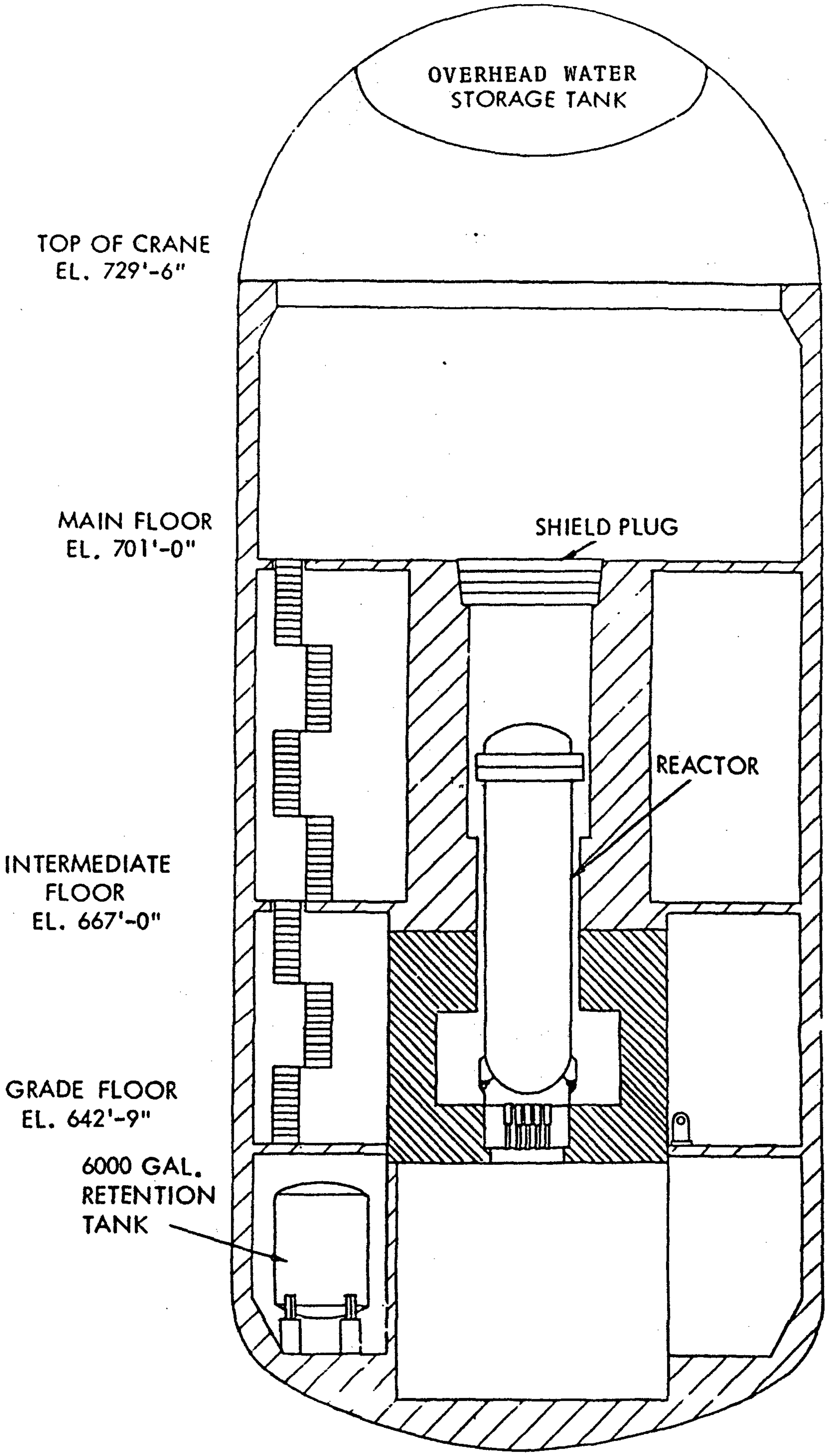
Wind Speed Frequency Distribution

FIGURE 3.6



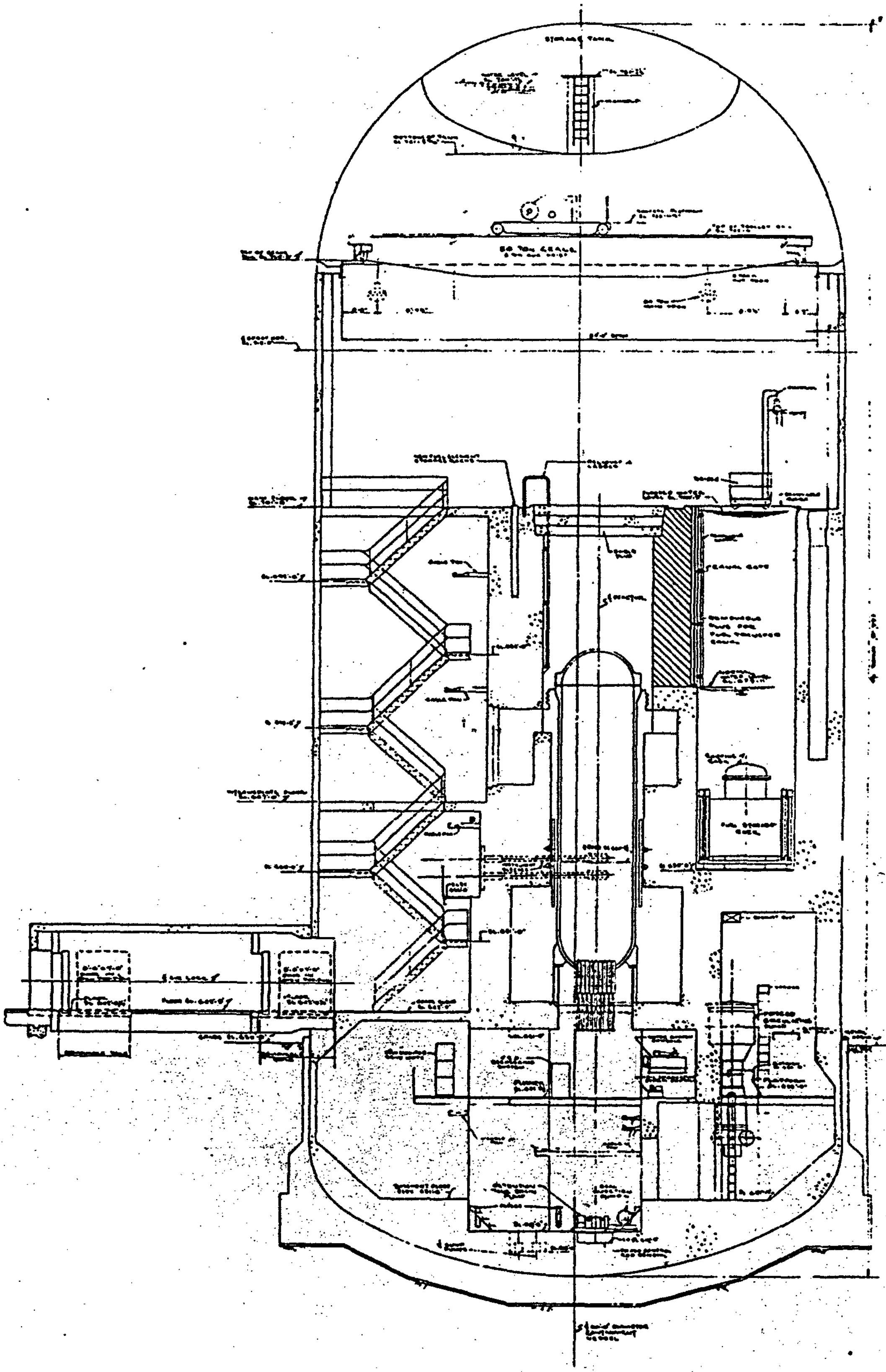
East Bank River Slope

FIGURE 3.7



<u>Containment Building Elevation</u>

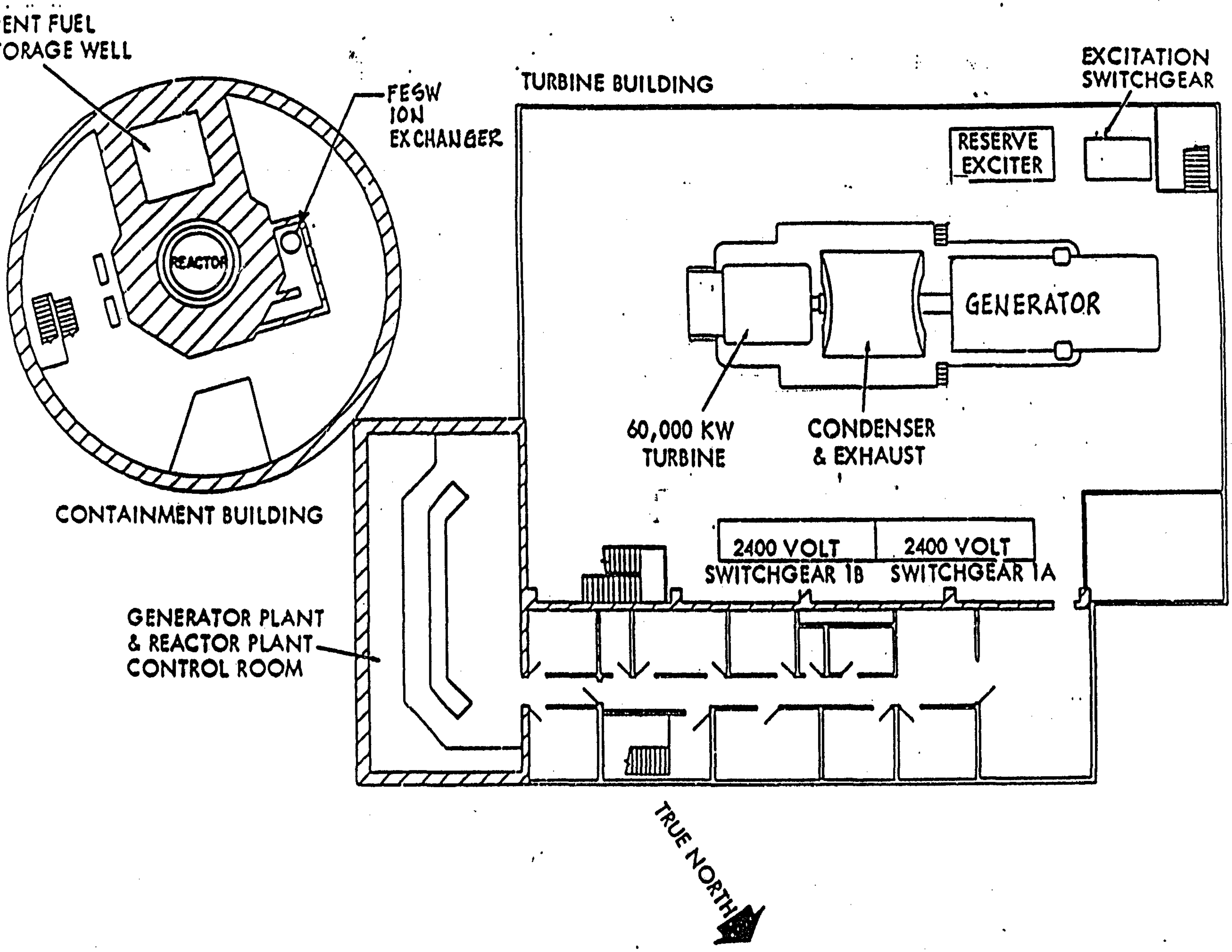
FIGURE 4.1



Containment Building General Arrangement

FIGURE 4.2

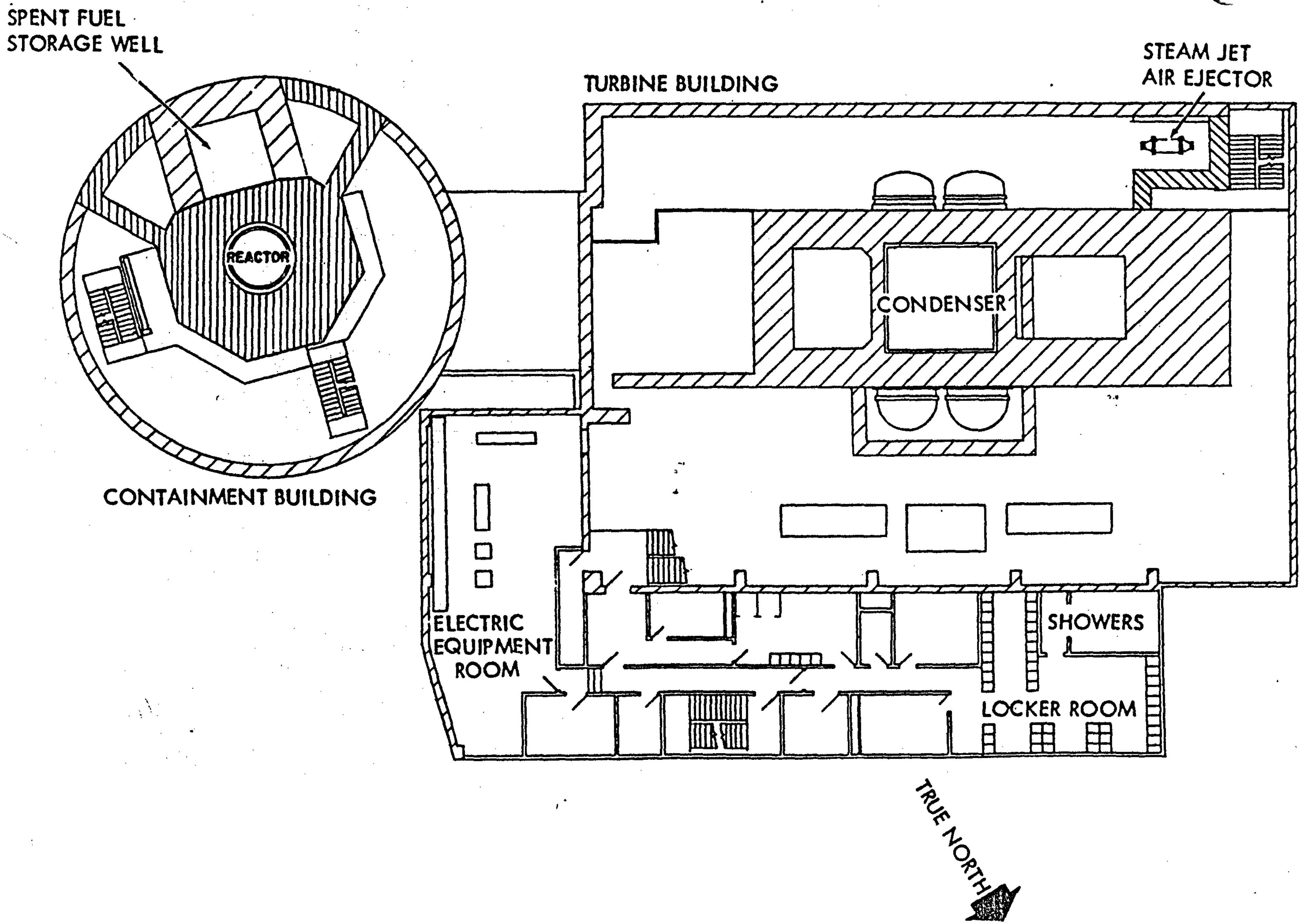
D-PLAN



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

FIGURE 4.3

D-PLAN



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

FIGURE 4.4

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.

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

(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.

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

All fuel assemblies and startup sources have been removed from the reactor core. The 29 control rods and other core components remain in the reactor vessel. The reactor vessel head is installed and partially bolted in place. The reactor vessel and primary systems have been drained to the maximum extent practical.

The LACBWR Reactor Pressure Vessel Removal Project was begun in August 2005. The reactor vessel with head installed, internals intact, and 29 control rods in place will be filled with low density cellular concrete. Attachments to the reactor vessel flange will be removed to a diameter of 119 inches. All other nozzles and appurtenances will be cut to within the diameter of the flange. Under-vessel nozzles and appurtenances will be removed from an envelope of within 6 inches of bottom dead center of the reactor vessel shell bottom.

The reactor pressure vessel will be removed from the Reactor Building and packaged for shipment by rail to the Barnwell Waste Management Facility in South Carolina for disposal by June 2007.

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 re-circulating 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 and attendant oil systems have been drained. The forced circulation pumps, auxiliary oil pumps, and hydraulic coupling oil pumps have been electrically disconnected and are not maintained operational.

All 16-inch and 20-inch forced circulation system piping will be filled with low density cellular concrete as part of the Reactor Pressure Vessel Removal Project begun in August 2005. Four 16-inch forced circulation inlet nozzles and four 16-inch outlet nozzles will be cut to allow removal of the reactor pressure vessel.

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.

The low pressure supply line allows the demineralized water from the Overhead Storage Tank, or the service water from the High Pressure Service Water (HPSW) supply line, to flow directly to the core spray header. 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 Reactor 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 1½-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

The Emergency Core Spray Pumps, associated piping and valves have been removed.

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. The ESWSS system has been removed.

System Status

The Alternate Core Spray System is not required to be operational in SAFSTOR. Therefore, the manual isolation valve to the Reactor Building is closed. The 6-inch supply line to the reactor pressure vessel head will be removed as interference to removal of the reactor pressure vessel. Motor-operated valves and instrumentation in the Turbine Building have been electrically removed. System components continue to serve requirements of the HPSW System. These components will be designated as part of the HPSW System in the near future.

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. A three-section aluminum cover, with two viewing windows per section, has been manufactured to cover the pool.

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 to the well, with either of two storage well pumps. The return line enters the top of the storage well and extends down to discharge at elevation 695 feet. The bottom inlet line ends at the biological shield wall and is sealed with a welded plug.

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 one of two "FESW Remote Operated Fill Valves," which are 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 Element Storage Well contains 333 irradiated fuel elements and will remain in operation as part of the SAFSTOR Program as long as wet fuel storage or wet fuel handling is necessary. Also stored in the well are 10 control rods, 2 antimony-beryllium startup sources, 24 stainless steel fuel element shroud cans, and 73 zirconium alloy fuel element shroud cans. These components will be removed, packaged and disposed of. Removal of irradiated hardware and other B&C wastes has been included in the scope of work during the Reactor Pressure Vessel Removal Project.

5.2.21 High Pressure Service Water System

The High Pressure Service Water (HPSW) system supplies fire suppression water and is available as backup cooling water for the Component Cooling Water coolers. During normal operation, HPSW system pressure is maintained by the LPSW system. A motor-driven HPSW pump with suction from the LPSW system is available for periods of high demand. With the motor-driven pump cycling in automatic, HPSW system pressure is maintained 110 to 135 psig at the expansion tank pressure switch elevation, 25 feet above site grade elevation. The pump is protected by a 35-psig low suction pressure trip. Backup supply is available from two HPSW diesel pumps. 1A HPSW Diesel Pump will start automatically if system pressure decreases to 90 psig. 1B HPSW Diesel pump will start automatically if system pressure decreases to 80 psig. The HPSW diesel pumps will maintain system pressure at approximately 150 psig. System pressure swings are cushioned by the air space in the HPSW surge tank.

The HPSW system is divided into two main loops. The internal loop serves the Turbine Building, Reactor Building, and Waste Treatment Building interior hose stations and sprinkler systems. The external loop supplies outside fire hydrants and Cribhouse sprinklers. The external loop is also cross-connected with the Fire Suppression System of the adjacent coal-fired generating facility, Genoa Unit 3. This cross-connect provides excess HPSW diesel pump capacity to this operating plant.

System Status

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 Unit 3, 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.

through a breaker on Turbine Building MCC 1A through a static switch in the inverter. Inverter 1C has been removed. Its distribution panel is powered from Turbine Building MCC 1A and has been renamed 1C 120-V AC Essential Power.

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 were divided into three separate and independent systems each with its own battery, battery charger, and distribution buses. The buses could be cross-connected but were normally isolated from each other. The Reactor Plant and Diesel Building batteries and chargers have been removed. The Generator Plant Battery and Charger remain as the sole sources of DC power to the 125-V DC distribution system. The once three separate systems have been interconnected by using installed bus tie breakers.

For the system, the Generator Plant Battery Charger provides the normal DC supply with the Generator Plant 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.

The Electrical Power Distribution System will be modified significantly. These modifications will configure the system to provide backup diesel generator power to the entire 480-V AC system. Facility power use has decreased below the capacity of the diesel generators making changes possible. Planned changes will ensure continued operation of normal lighting systems, air conditioning systems, and sanitary well water supply for habitability reasons and plumbing system use. Further simplification and reliability will be gained.

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 Reactor 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 the Reactor Building. Samples will be removed to the laboratory for analysis.

The Plant Manager is also responsible for the day-to-day activities of the operators, maintenance mechanics, instrument technicians and electricians. He 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. He is responsible for coordination of all Technical Specifications required tests.

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 Lead Mechanic 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. He is responsible for the preventive maintenance program established on those systems necessary to maintain the SAFSTOR condition. The Lead Mechanic 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 Lead Mechanic.

The Administrative Assistant is responsible for overall administration of LACBWR. She will maintain all records required under technical specifications for plant operation and will maintain a record of all activities of the SAFSTOR mode. The Administrative Assistant will ensure that all clerical functions are performed adequately. She will maintain all budget expense and project accounts and will coordinate preparation of the LACBWR budget. Duties will also include assigning to staff personnel required responses to regulatory agencies, other Dairyland departments, etc., and ensuring that these tasks are completed by the established deadline.

Additional administrative personnel will be made available to the Administrative Assistant as needed, and will assist in the clerical tasks at LACBWR. Such additional personnel will be qualified to perform required communication functions and will be assigned other tasks, as necessary, 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 Plant Manager 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. No more than a minority of the quorum shall have line responsibility for operation of the facility. The SRC shall meet at least once 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 Plant Manager. The Safety Review Committee and the Operations Review Committee will review all material as required by the Quality Assurance Program Description (QAPD) 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 governed by the requirements of 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 (HPT) Initial Training Program consists of the following topics:

a) <u>Science Training</u>

(1) Nuclear Theory

(2) Chemistry

(a) Non-radiological
(b) Radiochemistry

(3) Radiological Protection and Control (including surveys)

b) <u>Systems Training</u>

(1) Effluent Systems Sampling and Control

c) <u>Emergency Plan Training</u>

(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) <u>Environmental Program</u>

e) <u>Waste Disposal</u>

f) <u>Personnel Monitoring, including Internal Deposition Counting</u>

g) <u>Respiratory Protection Program</u>

h) <u>Radiation Monitoring and Instrumentation</u>

i) <u>Administrative Requirements</u>

j) <u>First Aid Training</u>

(f) Emergency Training

(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 ≥80% to pass exam.

6.4.3.4 Certified Fuel Handler Training Program. A training and certification program has been 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 provides 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.

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.

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 Manager 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.

Scheduled activities during SAFSTOR shall be performed within schedule intervals. A schedule interval is a time frame within which each scheduled activity shall be performed, with a maximum allowable extension not to exceed 25 percent of the schedule interval.

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. The LACBWR Decommissioning Plan was approved in August 1991, and the facility entered the SAFSTOR mode.

As discussed in Section 7.2, some modifications are considered beneficial to support the plant in the SAFSTOR condition.

Section 7.3.4 describes a major project undertaken at LACBWR. Duratek proposed to DPC in April 2005 that disposal of the Reactor Pressure Vessel (RPV) could proceed with fuel in the Reactor Building spent fuel pool. This disposal could occur prior to the Barnwell Waste Management Facility (BWMF) closing to out-of-compact waste in July 2008. In April 2005, DPC commissioned Duratek to study the feasibility of disposing of the RPV, intact, with existing internals at the BWMF. In August 2005, the results of this study led DPC to the decision to go forward with the actual removal of the RPV.

During the SAFSTOR period, DPC expects to ship the activated fuel to a federal repository, interim storage facility, or licensed temporary monitored retrievable storage facility. The timing of this action will be dependent on the availability of these facilities and their 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.

DPC is a part of the consortium of utilities that formed the Private Fuel Storage (PFS) Limited Liability Company (LLC) for the sole purpose of developing a temporary site for the storage of spent nuclear fuel for the industry. The Nuclear Regulatory Commission authorized the NRC staff to issue PFS a license on September 9, 2005. LACBWR spent fuel removal strategy and cask storage technology are being evaluated.

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 SAFSTOR FUNDING AND DECOMMISSIONING COST FINANCING

DPC is currently assuming a 30-50 year SAFSTOR period. 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. SAFSTOR and DECON costs are funded separately. SAFSTOR funding accommodates management of LACBWR spent fuel and provides assurance of continued funding through all modes of fuel storage prior to acceptance by the DOE. Mandated decommissioning funds will be available during the DECON period.

6.7.1 SAFSTOR

Pursuant to 10 CFR 50.54(bb), Dairyland Power Cooperative (DPC) has promulgated the following SAFSTOR spent fuel management and funding plan for LACBWR.

Independent of funding costs for SAFSTOR, DPC has established a Decommissioning Trust Fund and reports annually to the Nuclear Regulatory Commission the status of the fund. DPC understands that none of the funds in the Decommissioning Trust Fund may be used for spent fuel removal or for developing an Independent Spent Fuel Storage Facility (ISFSI). DPC has no plans to use any of the Decommissioning Trust Fund for an ISFSI or for spent fuel removal purposes.

DPC continues to fund the expense of SAFSTOR activities, including fuel storage costs, from the annual operating and maintenance budget. As part of generation expenses, SAFSTOR costs are recovered in rates that DPC charges distribution cooperative members under long-term, all requirements wholesale power contracts. DPC's rates to member cooperatives are annually submitted to the United States Rural Utilities Service (RUS) as part of RUS oversight of DPC operations. DPC is required by RUS lending covenants and RUS regulations to set rates at levels sufficient to recover costs and to meet certain financial performance covenants. DPC has always met those financial performance covenants and has satisfied the RUS regulations concerning submission and approval of its rates.

DPC's 25 member cooperatives set their own rates through participation in the DPC board of directors. The operations and maintenance budget approved by the DPC Board, and incorporated into rates submitted to and approved by the RUS, will be funded and available to pay SAFSTOR expenses as incurred.

DPC has found no need to separately fund SAFSTOR costs outside the regular operating and maintenance budget. SAFSTOR costs are relatively small compared to DPC's annual O&M costs for generation and transmission facilities, and DPC has continued the long-standing policy of recovering SAFSTOR costs as part of regular rates. DPC has seen no need to change the funding plan for SAFSTOR under those circumstances.

DPC continues to consider several alternatives to maintaining the LACBWR spent fuel in the current, wet-pool storage facility. If DPC decides to implement one of those alternatives, the funds for that alternative will be generated through DPC operating and maintenance budgets for the years when those activities will be undertaken. DPC does not intend to use any funds from the Decommissioning Trust Fund for those purposes.

DPC's annual budget for operating and maintenance activities at LACBWR accommodates SAFSTOR activities and includes funds for performing limited dismantlement at the LACBWR facility. Accomplishing limited dismantlement activities during SAFSTOR reduces the amount that will ultimately be necessary for decommissioning LACBWR after removal of the fuel. This approach takes advantage of the collective experience and familiarity of the LACBWR staff with the plant, and builds further conservatism into the funding plan for ultimate decommissioning of the facility.

6.7.2 DECON

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 original cost of the DECON phase was indicative of knowledge of technology as it existed 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.

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, was projected to be able to adequately fund the decommissioning cost in 2010, based on the original cost estimate of $20 million in 1983 dollars.

The decommissioning fund was 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 had accumulated to approximately $9,400,000. The decommissioning fund in the year 2000 was projected to reach $50 million (assumed equal to the original cost estimate), with the fund by the year 2010 at approximately $92,600,000 accrued.

The 1994 site-specific decommissioning cost study performed by Sargent & Lundy identified a need for increased funding. The Dairyland Power Cooperative Board of Directors authorized and approved an adjusted annual decommissioning accrual of $3 million with continued funding through 2010 to provide sufficient funding with commencement of decommissioning in 2019.

The cost study revision completed July 1998 placed the cost to complete decommissioning at $98.7 million in 1998 dollars. The annual decommissioning funding level required to meet the 2010 objective was $2.2 million. An adjustment to this level of funding was authorized by the Board of Directors.

A cost study update, prompted by significant changes in radioactive waste burial costs, as well as lessons learned on decontamination factors and methods, was prepared in November 2000. This update placed the cost to complete decommissioning at $79.2 million in Year 2000 dollars. During 2003, the cost study was revisited again to include changes in escalation rates, progress in limited dismantlement, and a revised reactor vessel weight definition. This update placed the cost to complete decommissioning at $79.5 million in Year 2003 dollars. Cooperative management believes that the balance in the nuclear decommissioning funds, together with future expected investment income on such funds, will be sufficient to meet all future decommissioning costs.

The DPC Board of Directors remains committed to assuring that adequate funding will be available for the decommissioning of the LACBWR facility and is prepared to adjust the funding level for the LACBWR Decommissioning Plan, from time to time, and/or take such other actions as it deems necessary or appropriate to provide such assurance, based upon its review of the most recent decommissioning cost estimate and other relevant developments in this area.

Every five 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.

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 President & CEO.

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 Reactor Building.

Additional small quantities of SNM are contained in neutron and calibration sources, 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

6.9.1 Fire Protection Plan

LACBWR can safely maintain and control the Fuel Element Storage Well in the case of the worst postulated fire in each area of the plant.

The fire protection plan at LACBWR is to prevent fire, effectively respond to fire, and to minimize the risk to the public from fire emergencies. The goals of the fire protection plan are fire prevention and fire protection. This fire protection plan, implemented through the fire protection program, provides defense-in-depth to fire emergencies and addresses the following objectives:

- **Prevent fires.** By administratively controlling ignition sources, flammable liquid inventory, and combustible material accumulation, fire risk is reduced. Welding and other hot work shall be performed only under Special Work Permit conditions and the use of a fire watch shall be required. Routine fire and safety inspections by LACBWR staff shall be conducted to ensure flammable liquids are properly stored and combustible material is removed. These inspections shall also require identification of fire hazards and result in action to reduce those hazards. General cleanliness and good housekeeping shall continue as an established practice and shall be checked during inspection.

- **Rapidly detect, control, and extinguish fires that do occur and could result in a radiological hazard.** Fire detection systems are installed to detect heat and smoke in spaces and areas of the protected premises of LACBWR. If fire detection systems or components are unavailable, increased monitoring of affected areas by personnel shall compensate for any loss of automatic detection. Fire barriers provide containment against the spread of fire between areas and provide protection to personnel responding to fire emergencies. Areas of high fire loading are provided with automatic reaction-type fire suppression systems or manually initiated fire suppression systems. These installed systems provide immediate fire suppression automatically or provide the means to extinguish fires without fire exposure to personnel manually initiating them. Manual fire extinguishing equipment is installed in all areas of the LACBWR facility. All fire protection equipment and systems are maintained, inspected, and tested in accordance with established guidelines. Compensatory actions and procedures for the impairment or unavailability of fire protection equipment are provided. A trained fire brigade, available at all times shall respond immediately to all fire emergencies. The function of the response by the fire brigade shall be to evaluate fire situations, to extinguish incipient stage fires, and to quickly realize the need for, and then summon, outside assistance. For any situation where a fire should progress beyond the incipient stage, qualified outside fire services shall provide assistance.

- **Minimize the risk to the public, environment, and plant personnel resulting from fire that could result in a release of radioactive materials.** Surface contamination has been reduced to minimal levels in most areas of the facility by cleanup efforts and the effects of long-term decay. Contamination surveys are performed routinely and areas identified for attention are deconned further. Good radiological work practices and contamination control are maintained. Radioactive waste generated is containerized and shipped for processing in accordance with approved procedures. Liquid effluents are collected in plant drain systems, processed, and monitored during discharge. Plant personnel are alerted to elevated radioactivity levels by area radiation monitors and air monitoring systems that are in operation at all times in buildings of the radiological controlled area. Gaseous and particulate air activities are continuously monitored prior to their release to the environment. Procedures and protocols exist to ensure risk is minimized to the public and members of the outside fire service.

6.9.2 Fire Protection Program

The fire protection program for the LACBWR facility is based on sound engineering practices and established standards. The function of the fire protection program is to provide the specific mechanisms by which the fire protection plan is implemented. The fire protection program utilizes an integrated system of administrative controls, equipment, personnel, tests, and inspections. Components of the fire protection program are:

6.9.2.1 Administrative Controls are the primary means by which the goal of fire prevention is accomplished. Administrative controls also ensure that fire protection program document content is maintained relevant to its fire protection function. By controlling ignition sources, combustible materials, and flammable liquids, and by maintaining good housekeeping practices, the probability of fire emergency is reduced. Procedures are routinely reviewed for adequacy and are revised as conditions warrant.

6.9.2.2 Fire Detection System. The LACBWR plant fire detection system is designed to provide heat and smoke detection. A Class B protected premises fire alarm system is installed which uses ionization or thermal-type fire detectors. Detectors cover areas throughout the plant and outlying buildings. The plant fire alarm system control panel is located in the Control Room. Alarms as a result of operation of a protection system or equipment, such as water flowing in a sprinkler system, the detection of smoke, or the detection of heat, are sounded in the Control Room. Alarm response is initiated from the Control Room.

The Administration Building fire detection system provides alarm functions using a combination of thermal detectors ionization detectors, and manual pull stations. Audible alarms are sounded throughout the building and provide immediate notice to occupants of fire emergency. The control panel for the Administration Building fire detection system is located within the Security Electrical Equipment Room.

6.9.2.3 Fire Barriers are those components of construction (walls, floors, and doors) that are rated in hours of resistance to fire by approving laboratories. Any openings or penetrations in these fire barriers shall be protected with seals or closures having a fire resistance rating equal to that of the barrier. The breaching of fire barriers is administratively controlled to ensure their fire safety function is maintained.

6.9.2.4 Fire Suppression Water System. The fire suppression water system is designed to provide a reliable supply of water for fire extinguishing purposes in quantities sufficient to satisfy the maximum possible demand. Fire suppression water is supplied by the High Pressure Service Water System (HPSW) which is normally pressurized from the Low Pressure Service Water (LPSW) system. Two HPSW diesel pumps provide fire suppression water when started manually or when started automatically by a decrease in HPSW pressure to <90 psig for HPSW Diesel Pump 1A or <80 psig for HPSW Diesel Pump 1B. Fire suppression water can be supplied from Genoa Unit 3 as a backup system to the HPSW system.

Fire suppression water is available from an external underground main at five 6-inch fire hydrants spaced at 200-foot intervals around the plant. Four outside hose cabinets contain the necessary hoses and equipment for hydrant operation.

Fire suppression water is available at five hose cabinets in the Turbine Building, one hose reel in the 1B Diesel Generator Building, and one hose cabinet in the Waste Treatment Building. Fire suppression water is available from hose reels located on each of four levels in the Reactor Building.

Fire suppression water is also supplied to sprinkler systems in areas with high fire loads. Sprinkler systems suppress fire in these areas without exposure to personnel. Automatic sprinkler systems are installed in the Oil Storage Room and in the Crib House HPSW diesel pump and fuel tank area. A manually initiated sprinkler system is installed in 1A Diesel Generator Room. An automatic reaction-type deluge system protects the Reserve Auxiliary Transformer located in the LACBWR switchyard.

6.9.2.5 Automatic Chemical Extinguishing Systems are installed in two areas of LACBWR containing high fire loads. The 1B Diesel Generator Room is protected by a CO_2 Flooding system. The Administration Building Records Storage Room is protected by a Halon system. These systems automatically extinguish fire using chemical agents, upon detection by their associated fire protection circuits. Fire in these areas is extinguished without exposure to personnel.

6.9.2.6 Portable Fire Extinguishers and Other Fire Protection Equipment. An assortment of dry chemical, CO_2, and Halon portable fire extinguishers rated for Class A, B, and C fires are located throughout all areas of the LACBWR facility. These extinguishers provide the means to immediately respond to incipient stage fires. Spare fire extinguishers are located on the Turbine Building grade floor.

Portable smoke ejectors are provided for the removal of smoke and ventilation of spaces. Smoke ejectors are located in the Change Room, on the Turbine Building mezzanine floor, and in the Maintenance Shop.

Four outside hose cabinets contain necessary lengths and sizes of fire hose for use with the yard fire hydrants. These hose cabinets also contain hose spanner and hydrant wrenches, nozzles, gate valves, coupling gaskets, and ball-valve wye reducers.

Tool kits are located in the Crib House outside fire cabinet and in the Maintenance Shop. Spare sprinkler heads and other sprinkler equipment is located in the Change Room locker. Rechargeable flashlights are wall-mounted in various locations and at entries to spaces. Portable radios are available at various locations and used for Fire Brigade communication.

6.9.2.7 The Fire Brigade is an integral part of the fire protection program. The Fire Brigade at LACBWR shall be organized and trained to perform incipient fire fighting duties. Personnel qualified to perform Operations Department duties and all LACBWR Security personnel shall be designated as Fire Brigade members and trained as such. Fire Brigade responsibilities shall be assigned to members of these groups while on duty.

The Fire Brigade shall be a minimum of two people at all times. The Duty Shift Supervisor (or his designee) shall respond to the fire scene as the Fire Brigade Leader. One member of the Security detail shall respond, as directed by the Fire Brigade Leader, and perform duties as the second Fire Brigade member.

The Control Room Operator shall communicate the status of fire detection system alarms or specific hazard information with the Fire Brigade, shall monitor and maintain fire header water pressure, and shall expeditiously summon outside fire service assistance as directed by the Fire Brigade Leader. The Control Room Operator shall use the page system to announce reports of fire, evacuation orders, and other information as requested by the Fire Brigade Leader.

6.9.2.8 Outside Fire Service Assistance. The LACBWR Fire Brigade is organized and trained as an incipient fire brigade. Fire Brigade Leaders are responsible for recognizing fire emergencies that progress beyond the limits of incipient stage fire fighting. Fire Brigade Leaders shall then immediately request assistance from outside fire services.

The LACBWR Emergency Plan contains a letter of agreement with the Genoa Fire Department. This letter of agreement states that the Genoa Fire Department is responsible for providing rescue and fire fighting support to LACBWR during emergencies. Upon request by the Genoa Fire Chief, all fire departments of Vernon County can be coordinated and directed to support the Genoa Fire Department during an emergency at LACBWR.

6.9.2.9 Reporting. Fire emergencies shall be documented under the following reporting guidelines:

- Any fire requiring Fire Brigade response shall be reported by the Duty Shift Supervisor using a LACBWR Incident Report.
- Any incident requiring outside fire service assistance within the LACBWR Site Enclosure (LSE fence) shall require activation of the Emergency Plan and shall require declaration of Unusual Event.

6.9.2.10 Training. Security badged visitors and contractors located at LACBWR shall receive indoctrination in the areas of fire reporting, plant evacuation routes, fire alarm response, and communications systems under General Employee Training.

Personnel who work routinely at LACBWR, and are given basic practical fire fighting instruction annually, are termed designated employees.

In addition to the annual practical fire fighting instruction, Fire Brigade members shall receive specific fire protection program instruction and participate in at least one drill annually.

Personnel not subject to Fire Brigade responsibilities shall receive training prior to performing fire watch duties.

6.9.2.11 Records. Fire Protection records shall be retained in accordance with Quality Assurance records requirements.

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 Plan, Safeguards Contingency Plan, Guard Force Training and Qualification Plan, and Security Control Procedures. The Security Plan for Transportation of LACBWR Hazardous Materials is found in the Process Control Program.

6.11 RECORDS

The Quality Assurance Program Description (QAPD) establishes measures for maintaining records which cover all documents and records associated with the decommissioning, operation, maintenance, repair, and modification of structures, systems, and components covered by the QAPD.

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.

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).
- 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.

7.3.4 Reactor Pressure Vessel Removal

Based on a feasibility study, DPC has entered into contract agreement with Duratek, Inc. for the removal and subsequent disposal of the intact Reactor Pressure Vessel (RPV) at the Barnwell Waste Management Facility in South Carolina. The RPV disposal is expected to be completed in the spring of 2007. DPC has included in the scope of work during this project removal of irradiated hardware and other B&C wastes.

The Decommissioning Plan discusses and provides for removal of unused equipment and plant system components, in accordance with 10 CFR 50.59, during SAFSTOR. The RPV removal is not specifically addressed in the decommissioning schedule. The removal of this large component, as defined in 10 CFR 50.2, is an activity requiring notice be made pursuant to 10 CFR 50.82, Termination of License, (a)(7). This notice was made by submittal to the NRC, in writing, on August 18, 2005. A copy of the submittal was sent to the affected State(s) before performing any decommissioning activity inconsistent with, or making any significant schedule change from, those actions and schedules described in the PSDAR, including changes that significantly increase the decommissioning cost.

7.3.5 Testing and Maintenance Program to Maintain Systems in Use

During the SAFSTOR period, a testing and maintenance program will continue for those systems 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.

LACBWR has established a program implementing the maintenance rule. This program contains key aspects of the maintenance rule. Included are those aspects specifically necessary to adequately identify structures, systems, and components (SSCs) to be monitored under the rule, establish goals and implement monitoring for those SSCs.

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 surrounding area, 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.

7.4.2 In-Plant Monitoring

Routine radiation dose rate and contamination surveys will be taken of plant areas along with more specific surveys needed to support activities 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) have been recalibrated to an equivalent Kr-85 energy. 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 in the ODCM.

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

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

7. DECOMMISSIONING ACTIVITIES - (cont'd)

7.4.4 Environmental Monitoring

Surrounding 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 the ODCM.

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

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. (Due to decay, as of October 2005 only 31.3% of the Kr-85 activity remains - 135 Curies.)

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 two (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 Reactor Building. The next case is the dose due to a ground level release at the Exclusion Area Boundary. The maximum dose at the 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

$$1839 \frac{\text{mRem/hr}}{\text{Ci/m}^3} \times 1.38\text{ E-5 Ci/m}^3 \times 2\text{ hr} = 0.05\text{ mRem} \quad (\text{as of } 10/05 = 0.02\text{ mRem})$$

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

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

Skin Dose at 500m E

$$1.53\ \mathrm{E5} \frac{\mathrm{mRem/hr}}{\mathrm{Ci/m}^3} \, x \, 1.38E\text{-}5\ \mathrm{Ci/m}^3 \ x \ 2\,\mathrm{hr} = 4.2\ \mathrm{mRem}\ (\text{as of } 10/05 = 1.3\ \mathrm{mRem})$$

Ground Level Release at EAB

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

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

Whole Body Dose at 338m	Skin Dose at 339m
10/87 = 0.49 mRem	10/87 = 40.4 mRem
10/05 = 0.15 mRem	10/05 = 12.7 mRem

Ground Level Release at Emergency Planning Zone Boundary

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

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

Whole Body Dose at 100m E	Skin Dose at 100m E
10/87 = 2.25 mRem	10/87 = 187 mRem
10/05 = 0.70 mRem	10/05 = 58.5 mRem

As can be seen, the estimated maximum whole body dose is more than a factor of 30,000 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 (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. (Due to decay, as of Oct. 2005 only 31.3% of the Kr-85 activity remains – 11,193 Curies.)

Elevated Release

Whole Body Dose at 500m E	Skin Dose at 500m E
10/87 = 4.2 mRem	10/87 = 350 mRem
10/05 = 1.3 mRem	10/05 = 109.6 mRem

Ground Level Release at EAB

Whole Body Dose at 338m	Skin Dose at 338m
10/87 = 40.2 mRem	10/87 = 3.34 Rem
10/05 = 12.6 mRem	10/05 = 1.05 Rem

Ground Level Release at Emergency Planning Zone Boundary

Whole Body Dose at 100m E	Skin Dose at 100m E
10/87 = 186 mRem	10/87 = 15.6 Rem
10/05 = 58.2 mRem	10/05 = 4.9 Rem

As can be seen, the estimated maximum whole body dose is more than a factor of 400 below the 10 CFR 100 dose limit of 25 Rem (25,000 mRem) to the whole body within a 2-hour period.

10.5 CERTIFICATION

Upon successful completion of the initial certification training program, the Plant Manager 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 two 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.

LAC-TR-138

LACBWR

INITIAL

SITE CHARACTERIZATION SURVEY

FOR SAFSTOR

By:

Larry Nelson
Health and Safety Supervisor

October 1995

Revised: November 2005

Dairyland Power Cooperative
3200 East Avenue South
La Crosse, WI 54601

ATTACHMENT 1

SPENT FUEL RADIOACTIVITY INVENTORY

Decay-Corrected to October 2005

Radionuclide	*Half Life (Years)*	*Activity (Curies)*	*Radionuclide*	*Half Life (Years)*	*(Curies)*
Ce-144	7.801 E-1	0.374	Sr-90	2.770 E + 1	7.36E5
Cs-137	3.014 E+1	1.11E6	Pu-241	1.440 E+1	4.84E5
Ru-106	1.008 E+0	7.64	Fe-55	2.700 E+0	5.52E3
Cs-134	2.070 E+0	864	Ni-59	8.000 E+4	287
Kr-85	1.072 E+1	3.68E4	Tc-99	2.120 E+5	276
Ag-110m	6.990 E-1	2.32E-3	Sb-125	2.760 E+0	3.17
Co-60	5.270 E+0	6.20E3	Eu-155	4.960 E+0	14.1
Pm-147	2.620 E+0	377	U-234	2.440 E+5	63.7
Ni-63	1.000 E+2	3.13E4	Am-243	7.380 E+3	61
Am-241	4.329 E+2	1.43E4	Cd-113m	1.359 E+1	7.20
Pu-238	8.774 E+1	1.10E4	Nb-94	2.000 E+4	15.9
Pu-239	2.410 E+4	8.83E3	Cs-135	3.000 E+6	14.0
Pu-240	6.550 E+3	7.15E3	U-238	4.470 E+9	12.2
Eu-154	8.750 E+0	986	Pu-242	3.760 E+5	8.58
Cm-244	1.812 E+1	1.83E3	U-236	2.340 E+7	6.32
H-3	1.226 E+1	202	Sn-121m	7.600 E+1	3.78
Eu-152	1.360 E+1	207	Np-237	2.140 E+6	2.19
Am-242m	1.505 E+2	452	U-235	7.040 E+8	1.89
			Sm-151	9.316 E+1	1.32
			Sn-126	1.000 E+5	0.7
			Se-79	6.500 E+4	0.552
			I-129	1.570 E+7	0.39
			Zr-93	1.500 E+6	0.111

Total Activity = 2.46 E6 Curies

ATTACHMENT 2

CORE INTERNAL/RX COMPONENT RADIONUCLIDE INVENTORY – OCTOBER 2005

Components	Estimated Curie Content				
	Co-60	Fe-55	Ni-63	Other Nuclides $T_{1/2} > 5y$	Total
In Reactor					
Fuel Shrouds (72 Zr, 8 SS)	2,142	664	1,196	8	4,010
Control Rods (29)	473	51	722	8	1,254
Core Vertical Posts (52)	123	6	56	2	187
Core Lateral Support Structure	883	226	681	4	1,794
Steam Separators (16)	3,240	828	2,499	15	6,582
Thermal Shield	140	36	109	0.5	286
Pressure Vessel	34	11	9	--	54
Core Support Structure	626	160	483	3	1,272
Horizontal Grid Bars (7)	17	4	13	--	34
Incore Monitor Guide Tubes	30	2	540	3	575
Total	7,708	1,988	6,308	43.5	16,048
In FESW					
Fuel Shrouds (24 SS)	1,324	157	2,108	13	3,602
Fuel Shrouds (73 Zr)	89	11	84	2	186
Control Rods (10)	335	25	805	9	1,174
Start-up Sources (2)	308	24	138	2	472
Total	2,056	217	3,135	26	5,434

ATTACHMENT 3

PLANT SYSTEMS INTERNAL RADIONUCLIDE INVENTORY - OCTOBER 2005

Plant System	Nuclide Activity, in µCi				System Total µCi Content
	Fe-55	Alpha	Co-60	Cs-137	
CB Ventilation	17	--	155	113	285
Offgas - upstream of filters	*SYSTEM*	*REMOVED*			
Offgas - downstream of filters	*SYSTEM*	*REMOVED*			
TB drains	179	40	1,647	3,323	5,189
CB drains	399	3	3,682	1,595	5,679
TB Waste Water	38	7	349	80	474
CB Waste Water	2,206	79	20,348	1,528	24,161
Main Steam	2,731	290	25,193		28,214
Turbine	10	2	90	133	235
Primary Purification	935	12	8,624		9,571
Emergency Core Spray	*SYSTEM*	*REMOVED*			
Overhead Storage Tank	137	34	1,260	518	1,949
Seal Inject	17	4	155	37	213

ATTACHMENT 3

PLANT SYSTEMS INTERNAL RADIONUCLIDE INVENTORY – OCTOBER 2005 - (cont'd)

Plant System	Nuclide Activity, in µCi				System Total µCi Content
	Fe-55	Alpha	Co-60	Mn-54	
Decay Heat	1,051	490	9,690		11,231
Boron Inject	*SYSTEM*	*REMOVED*			
Reactor Coolant PASS	*SYSTEM*	*REMOVED*			
Alternate Core Spray	210	94	1,938		2,242
Shutdown Condenser	*SYSTEM*	*REMOVED*			
Control Rod Drive Effluent	1,576	720	14,535		16,831
Forced Circulation	15,759	7,000	145,345	0.2	152,345
Reactor Vessel and Internals	26,264	12,000	242,242	0.4	280,506
Condensate after beds & Feedwater	*SYSTEM*	*REMOVED*			
Condensate to beds	*SYSTEM*	*REMOVED*			

ATTACHMENT 3

PLANT SYSTEMS INTERNAL RADIONUCLIDE INVENTORY – OCTOBER 2005 - (cont'd)

Plant System	Nuclide Activity, in µCi					System Total µCi Content
	Fe-55	Alpha	Co-60	Mn-54	Cs-137	
Fuel Element Storage Well System	8,930	390	82,362			91,682
Fuel Element Storage Well - all but floor	14	5	126		3,057	3,202
Fuel Element Storage Well floor	273,149	7,600	2,519,318	0.3	27,246	2,827,313
Resin lines	1,366	100	12,597			14,063
Main Condenser	115,563	8,500	1,065,865	2		1,189,930