DAIRYLAND POWER
COOPERATIVE

December 26, 2007

In reply, please refer to LAC-14020

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:two
Enclosures

cc: Kris Banovac
NRC Project Mgr.

NMSS01
FSME

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2007 LACBWR Decommissioning Plan Review

NOTE:	***Changes described following are as they appear in 2007 D-Plan pages as revised.***
Cover Page	Update revision date.
Page 0-1	Table of Contents: Page is issued with minor format changes.
Page 0-2	Table of Contents: Section 6, "Decommissioning Program," has been revised by addition of two sections, 6.11 and 6.12, previously contained in Section 7, "Decommissioning Activities." These two sections are added as being more applicable to the Decommissioning Program description. "Records" section is renumbered to 6.13. Page numbering is updated with content addition. Section 6 changes are: ***6.11*** ***Testing and Maintenance of SAFSTOR Systems*** (formerly Section 7.3.5, "Testing and Maintenance Program to Maintain Systems in Use") ***6.12*** ***Plant Monitoring Program*** (formerly Section 7.4, same title) ***6.13*** Records Table of Contents: Section 7, "Decommissioning Activities," has been revised by title changes and content addition. Major subheading 7.3, "Activities during SAFSTOR Period," has been deleted as being redundant to title of Section 7, "Decommissioning Activities." Page numbering is updated with content addition. Section 7 changes are: *7.3* ***Significant SAFSTOR Licensing Actions*** (formerly Section 7.3.1) *7.4* ***Area and System Decontamination*** (formerly Section 7.3.2) *7.5* ***Removal of Unused Equipment During SAFSTOR*** (formerly Section 7.3.3) *7.6* ***Reactor Pressure Vessel Removal*** (formerly Section 7.3.4) *7.7* ***Environmental Impact*** (new section, new content)
Page 0-3	Table of Contents: Section 8, "Health Physics," page numbering is updated with content addition in Section 8.2, "ALARA Program."
Page 0-4	List of Figures: Two figures are added depicting structural changes in the Reactor Building for reactor pressure vessel (RPV) removal. Schedule for LACBWR has been revised and name shortened. Changes to the List of Figures are: ***Figure 4.6*** ***Reactor Building Opening*** ***Figure 4.7*** ***Reactor Building Bi-Parting Door*** ***Figure 6.2*** ***LACBWR Schedule***
Page 2-4	Section 2.5.1, Failed Fuel: In final paragraph of section continuing at top of page, reference to transuranic contaminants being located in the Reactor Vessel is deleted. RPV removal and disposal has been completed.

Page 2-4	Section 2.5.2, Fuel Element Storage Well Leakage: Statement that ***FESW leakage has stabilized over the years to an average of approximately 21 gallons per day*** is added to discussion.
Page 3-7	Section 3.4.7, Ultimate Heat Sink and Low Flow Conditions: Final paragraph referring to a test documented in Technical Report 137 has been replaced with a more appropriate general discussion of the minimal consequences associated with loss of spent fuel cooling. Reader is directed to information contained in ***Section 9, "SAFSTOR Accident Analysis,"*** which more fully presents the information.
Page 4-1	Section 4.2.1, Reactor Building: In third paragraph at bottom of page verbs are changed to past tense to describe original Reactor Building (RB) design characteristics. RB has been modified for RPV removal.
Page 4-2	Section 4.2.1, Reactor Building: In fifth paragraph, third sentence is revised to state, *"The airlock is 21 ft. 6 in. long between its two **rectangular doors that measure** 5 ft. 6 in. by 7 ft."* The description that the doors are large enough to permit passage of a spent fuel element shipping cask has been deleted. Airlock openings are not big enough for today's standard casks. In sixth paragraph, RB freight door description is revised to comment that freight door, having never been used, ***was intended to accommodate*** large pieces of equipment. New content is also added at paragraph seven to describe RB opening and bi-parting door installed for RPV removal.
Page 4-3	Section 4.2.1, Reactor Building: In second paragraph description of overhead storage tank (OHST) is revised following RPV removal. OHST will be used in fuel handling operations, but exact configuration and design of cask pool/tank to be installed in bio-shield cavity where RPV resided is yet to be determined.
Page 4-4 And Page 4-5	Pages are issued due to content shift with no other changes.
Section 4 Figures	***Figure 4.6 and Figure 4.7*** are added to the section depicting the RB opening and RB bi-parting door.
Page 5-3	Section 5.2.1, Reactor Vessel and Internals: System description is revised by changing verbs to past tense. System Status is revised to reflect removal of the RPV.
Page 5-4	Section 5.2.2, Forced Circulation System: System description is revised by changing verbs to past tense. System Status is revised to reflect removal of the RPV and removal of system piping in lower cavity.

Page 5-41 Through Page 5-43	Section 5.4.2, System Radiation Levels: In the listing of survey point dose rates, current dose rates are updated to 2007 values and reflect equipment removal.
Page 6-1	Section 6.2, Organization and Responsibilities: In second paragraph, ***Quality Assurance Program Description (QAPD)*** is corrected to more appropriate title with acronym added.
Page 6-3	Section 6.2, Organization and Responsibilities: In second paragraph, title of Lead Mechanic is removed. DPC no longer utilizes the "lead" classification in its crafts organization. Description is revised to reflect responsibilities of LACBWR ***Maintenance Mechanics*** who continue to perform tasks by direction of the Plant Manager. Reporting relationships remain unchanged.
Page 6-4	Section 6.2, Organization and Responsibilities: In final paragraph of section, reference to Quality Assurance Program Description (QAPD) is shortened to simply ***QAPD.***
	Section 6.3, Contractor Use: Section is revised by condensing content to more affirmatively describe recent and expected contractor employment. Requirements of the ***QAPD*** will continue to govern the use of contractors at LACBWR.
Page 6-7	Section 6.4.3.2, describing the Health Physics Technician (HP) Continuing Training Program, at item g) (1), *intralaboratory comparisons in analytical chemistry--crosscheck analysis* has been revised to state, *intralaboratory comparisons in* ***radio-****chemistry (crosscheck analysis)* will be covered. Change more appropriately describes analyses performed at a nuclear facility.
Page 6-10 And Page 6-11	Section 6.6, Schedule: Entire section is revised to provide a better depiction of the schedule of activities, past and future, at LACBWR. B/C waste and RPV removal have been completed. Funding for decommissioning is projected to be sufficient. An ISFSI project at the LACBWR site has commenced and plans for license termination are being discussed.
Page 6-12	***Section 6.7.1, SAFSTOR Funding:*** Title of section is renamed from "SAFSTOR" with no other changes.
Page 6-13 And Page 6-14	***Section 6.7.2, Decommissioning Cost Financing:*** Title of section is renamed from "DECON." Entire section is revised to provide a better depiction of LACBWR decommissioning funding. Dated material is removed and discussion is improved to provide clarity to basis and current status of decommissioning funding. DPC commitment to assure adequate funding is reaffirmed.
Page 6-15 Through Page 6-18	Pages are issued due to content shift with no other changes.

Page 6-19	***Section 6.11, Testing and Maintenance of SAFSTOR Systems:*** Section is moved from Section 7.3.5 and renamed from "Testing and Maintenance Program to Maintain Systems in Use," with minor changes to content. Change is performed to include information more applicable to Decommissioning Program that was previously contained in Decommissioning Activities.
Page 6-19 And Page 6-20	***Section 6.12, Plant Monitoring Program:*** Section is moved from Section 7.4 with no other changes to content. Change is performed to include information more applicable to Decommissioning Program that was previously contained in Decommissioning Activities.
Page 6-21	Section *6.13*, Records: With movement of Section 7 content to Sections 6.11 and 6.12 described previously, section is renumbered with no other changes.
Figure 6.2	***LACBWR Schedule:*** Figure 6.2 is revised to provide a better depiction of the schedule for LACBWR as it develops. Title is shortened from "Tentative Schedule for LACBWR Decommissioning."
Page 7-1 And Page 7-2	***Section 7.3, Significant SAFSTOR Licensing Activities:*** Section is renumbered with deletion of "Activities during SAFSTOR Period," title which is redundant to title of Section 7. Editorial changes are made in first, second, and fourth paragraphs of section to correct terms and provide clarification.
Page 7-2	***Section 7.4, Area and System Decontamination:*** Section is renumbered with no other changes.
Page 7-3	***Section 7.5, Removal of Unused Equipment during SAFSTOR:*** Section is renumbered with no other changes.
Page 7-3 Through Page 7-10	***Section 7.6, Reactor Pressure Vessel Removal*:** Section is renumbered and revised to provide full description of RPV removal activities. ***Section 7.6.1*** is added to describe temporary lifting device installed and used for RPV removal. All TLD equipment was removed following RPV removal with the exception of two rocker bearing assemblies installed on the bio-shield at elevation 701' and two bearing assemblies mounted at the Reactor Building wall opening. ***Section 7.6.2*** is added to provide information addressing NUREG-0612 compliance during RPV removal activities. ***Section 7.6.3*** is added to provide information concerning RPV removal facility changes and 50.59 Evaluations performed during the project. ***Section 7.6.4*** is added to incorporate and provide reference to all analyses completed in support of RPV removal 50.59 Evaluation conclusions.
Page 7-10	***Section 7.7, B/C Waste Removal:*** Section with new content is added describing completed B/C waste removal activities.

Page 7-10 And Page 7-11	***Section 7.8, Environment Impact:*** Section with new content is added describing environmental impact of dismantlement activities at LACBWR to date. Impact of activities has been previously evaluated. Activities have created no new environmental impacts. Evaluation of environmental impact will occur during License Termination Plan completion.
Page 8-2 Through Page 8-4	Section 8.2.2, Application of ALARA: Section is revised to provide information and changes previously implemented that bolster the ALARA program and practices at LACBWR. More restrictive dose limits are added. Restrictions and terms are clarified.
Page 8-5 Through Page 8-12	Remainder of Section 8 pages are issued due to content shift with no other changes.
Page 9-2 And Page 9-3	Section 9.2, Spent Fuel Handling Accident: The curie content remaining as of *October 2006* and calculated values for Whole Body Dose and Skin Dose as of *October 2006* are updated to ***October 2007.***
Page 9-4	Section 9.3, Shipping Cask or Heavy Load Drop into FESW: The curie content remaining as of *October 2006* and calculated values for Whole Body Dose and Skin Dose as of *October 2006* are updated to ***October 2007.***

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

Cover Page	Update revision date.
Page 2	Section 2.0 (2), Fuel Element Storage Well Leakage: Two statements are added to first paragraph to reflect D-Plan content; a third is added to comment on FESW leakage deposition: ***"In 1993, the FESW pump seals were discovered to be defective and were replaced, which reduced the leak rate to approximately 1 gph. FESW leakage has stabilized over the years to an average of approximately 21 gallons per day. This leakage has been contained within the steel shell of the Reactor Building."*** Second paragraph discussing FESW leakage through structural concrete is deleted because it creates confusion as to potential problems caused by leakage within the Reactor Building. As stated in this revision, leakage is contained within the steel shell. Demolition of structures, including the Reactor Building, is planned for final decommissioning of LACBWR.
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 2007,*** replacing pages that had been decay-corrected to *October 2006.*
Page 25	Curie content is deleted at *"In Reactor"* section of table and is replaced by note stating, ***"Reactor Vessel was processed, packaged and disposed of in 2007."***

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 12/21/2007

SUBJECT: Changes to LACBWR Controlling Documents

I. The following documents have been revised:

DECOMMISSIONING PLAN, revised December 2007

Remove and replace the following pages:

Cover Page
Pages 0-1 thru 0-4,
Pages 2-4, 3-7, 4-1 thru 4-5
Figures 4.6 and 4.7
Pages 5-3 thru 5-4 and 5-41 thru 5-43
Pages 6-1, 6-3, 6-4, 6-7, 6-10 thru 6-21
Figures 6.2
Pages 7-1 thru 7-11
Pages 8-2 thru 8-12
Pages 9-2 thru 9-4

SITE CHARACTERIZATION SURVEY

Remove and replace the following pages:

Cover Page
Page 2
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.

LA CROSSE BOILING WATER REACTOR

(LACBWR)

DECOMMISSIONING

PLAN

Revised
December 2007

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

TABLE OF CONTENTS

		Page No.
1.	**Introduction**	1-1
1.1	Selection of SAFSTOR	1-1
1.2	References	1-3
2.	**La Crosse Boiling Water Reactor Operating History**	2-1
2.1	Introduction	2-1
2.2	Initial Construction and Licensing History	2-1
2.3	Operating Record	2-2
2.4	Decision for Shutdown	2-3
2.5	Operating Events which Affect Decommissioning	2-3
3.	**Facility Site Characteristics**	3-1
3.1	Geography and Demography Characteristics	3-1
3.2	Transportation, Industrial and Military Facilities within Proximity to the Plant	3-1
3.3	Meteorology	3-2
3.4	Hydrology	3-4
3.5	Geology, Seismology and Geotechnical Engineering	3-8
3.6	References	3-15
4.	**Facility Description**	4-1
4.1	General Plant Description	4-1
4.2	Buildings and Structures	4-1
5.	**Plant Status**	5-1
5.1	Fuel Inventory	5-1
5.2	Plant Systems and their Status	5-3
5.3	Radionuclide Inventory Estimates	5-39

TABLE OF CONTENTS

		Page No.
5.4	Radiation Levels	5-40
5.5	Plant Personnel Dose Estimate	5-43
5.6	Sources	5-44
5.7	Radiation Monitoring Instrumentation	5-44
6.	**Decommissioning Program**	6-1
6.1	Objectives	6-1
6.2	Organization and Responsibilities	6-1
6.3	Contractor Use	6-4
6.4	Training Program	6-5
6.5	Quality Assurance	6-10
6.6	Schedule	6-10
6.7	SAFSTOR Funding and Decommissioning Cost Financing	6-11
6.8	Special Nuclear Material (SNM) Accountability	6-14
6.9	SAFSTOR Fire Protection Program	6-14
6.10	Security During SAFSTOR and/or Decommissioning	6-19
6.11	Testing and Maintenance of SAFSTOR Systems	6-19
6.12	Plant Monitoring Program	6-19
6.13	Records	6-21
7.	**Decommissioning Activities**	7-1
7.1	Preparation for SAFSTOR	7-1
7.2	SAFSTOR Modifications	7-1
7.3	Significant SAFSTOR Licensing Actions	7-1
7.4	Area and System Decontamination	7-2
7.5	Removal of Unused Equipment During SAFSTOR	7-3
7.6	Reactor Pressure Vessel Removal	7-3
7.7	B/C Waste Removal	7-10
7.7	Environmental Impact	7-10

TABLE OF CONTENTS

		Page No.
8.	**Health Physics**	8-1
8.1	Organization and Responsibilities	8-1
8.2	ALARA Program	8-1
8.3	Radiation Protection Program	8-5
8.4	Radiation Monitoring	8-7
8.5	Radiation Protection Equipment and Instrumentation	8-10
8.6	Radioactive Waste Handling and Disposal	8-11
8.7	Records	8-12
8.8	Industrial Health and Safety	8-12
9.	**SAFSTOR Accident Analysis**	9-1
9.1	Introduction	9-1
9.2	Spent Fuel Handling Accident	9-1
9.3	Shipping Cask or Heavy Load Drop into FESW	9-4
9.4	Loss of FESW Cooling	9-5
9.5	FESW Pipe Break	9-5
9.6	Uncontrolled Waste Water Discharge	9-6
9.7	Loss of Offsite Power	9-6
9.8	Seismic Event	9-7
9.9	Wind and Tornado	9-8
9.10	References	9-8
10	**SAFSTOR Operator Training and Certification Program**	10-1
10.1	Introduction	10-1
10.2	Applicability	10-1
10.3	Initial Certification	10-1
10.4	Proficiency Training and Testing	10-2
10.5	Certification	10-3
10.6	Physical Requirements	10-3
10.7	Documentation	10-3

TABLE OF CONTENTS

LIST OF FIGURES

Figure 3.1	General Site Location Map
Figure 3.2	Population Dispersion
Figure 3.3	Effluent Release Boundary
Figure 3.4	Genoa Site Map
Figure 3.5	Monthly Average Meteorological Data
Figure 3.6	Wind Speed Frequency Distribution
Figure 3.7	East Bank River Slope
Figure 3.8	Generalized Soil Profile
Figure 4.1	Containment Building Elevation
Figure 4.2	Containment Building General Arrangement
Figure 4.3	Main Floor of Turbine Building, El. 668'0"
Figure 4.4	Mezzanine Floor of Turbine Building, El. 654'0"
Figure 4.5	Grade Floor of Turbine Building, El. 640'0"
Figure 4.6	Reactor Building Opening
Figure 4.7	Reactor Building Bi-Parting Door
Figure 6.1	LACBWR Organization Chart
Figure 6.2	LACBWR Schedule

LIST OF TABLES

		Page No.
Table 3-1	Wind Direction Frequency Distribution at LACBWR Site and La Crosse NWS	3-3
Table 5-1	Spent Fuel Radioactivity Inventory	5-1

Therefore, extra precautions will be taken in monitoring for and containing any fission product and transuranic radionuclide contaminants during the eventual disassembly of the above listed systems. The majority of this material is located on horizontal surfaces in the Fuel Element Storage Well.

2.5.2 Fuel Element Storage Well Leakage

The stainless steel liner in the Fuel Element Storage Well (FESW) has had a history of leakage. From the date of initial service until 1980, the leakage increased from approximately 2 gallons per hour (gph) to just over 14 gph. In 1980, epoxy was injected behind the liner and leakage was reduced to approximately 2 gph. In 1993, the FESW pump seals were discovered to be defective and were replaced, which reduced the leak rate to approximately 1 gph. FESW leakage has stabilized over the years to an average of approximately 21 gallons per day. FESW water level is continuously monitored in the control room and verified periodically by local inspection. The control room level instrument(s) generate an audible alarm when FESW level decreases to a selected level which is significantly above the minimum allowable level as specified in the technical specifications.

2.5.3 References

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

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

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

At 639 feet MSL, a flood emergency is declared. At 643 feet MSL a flood crisis level is declared. At this point, actions are taken to minimize the differential pressure on the Reactor Building. The warning available to the facility of flood cresting is 4-5 days following crest at Minneapolis, Minnesota.

3.4.7 Ultimate Heat Sink and Low Flow Conditions

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

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

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

As described in Section 9, "SAFSTOR Accident Analysis," substantial time is available for restoration of spent fuel cooling. If water is added to the FESW, any consequences of water heat up can be delayed or prevented.

3.4.8 Ground Water

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

4. FACILITY DESCRIPTION

4.1 GENERAL PLANT DESCRIPTION

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

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

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

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

4.2 BUILDING AND STRUCTURES

4.2.1 Reactor Building

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

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

The Reactor Building was designed to withstand the instantaneous release of all the energy of the primary system to the Reactor Building atmosphere at an initial ambient temperature of 80°F, neglecting the heat losses from the building and heat absorption by internal structures. The design pressure was 52 psig, compared to a calculated maximum pressure buildup of 48.5 psig following the maximum credible accident while in operation. The Reactor Building shell was designed and constructed according to the ASME Boiler and Pressure Vessel Code, Sections II, VIII, and IX, and Nuclear Code Cases 127ON, 1271N, and 1272N.

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

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

The containment bottom head above el. 621 ft. 6 in. and the shell cylinder from the bottom head to approximately 9 in. above grade elevation (639 ft. 9 in.) are enveloped by reinforced concrete laid over a 1/2 in. thickness of premolded expansion joint filler. The reinforced concrete consists of a lower ring, mating with the pile capping concrete. The ring is approximately 4½ ft. thick at its bottom and 2½ ft. thick at a point 1½ ft. below its top (due to inner surface concavity). The ring then tapers externally to a thickness of 9 in. at the top (el. 627 ft. 6 in.) and the 9 in. thickness of concrete extends up the wall of the shell cylinder to 639 ft. 9 in. The filler and concrete are not used, however, where cavities containing piping and process equipment are immediately adjacent to the shell.

Except for areas of the shell adjacent to other enclosures, the exterior surface of the shell above el. 639 ft. 9 in. is covered with 1½-inch-thick siliceous fiber insulation, faced with aluminum. The insulation of the dome is Johns-Manville Spintex of 9 lb/ft^3 density, faced with embossed aluminum sheet approximately 0.032 in. thick. The insulation of the vertical walls is Johns-Manville Spintex of 6 lb/ft^3 density, faced with corrugated embossed aluminum sheet approximately 0.016 in. thick. The insulation minimizes heat losses from the building and maintains the required metal temperature during cold weather, and reduces the summer air-conditioning load.

The shell includes two airlocks. The principal access to the shell is through the personnel airlock that connects the Reactor Building to the Turbine Building. The airlock is 21 ft. 6 in. long between its two rectangular doors that measure 5 ft. 6 in. by 7 ft. The Reactor Building can also be exited, if necessary, through the emergency airlock, which is 7 ft. long and 5 ft. in diameter, with two circular doors of 32½ in. diameter (with a 30-in. opening). Both airlocks are at el. 642 ft. 9 in. and lead to platform structures from which descent to grade level can be made.

An 8 ft. by 10 ft. freight door opening in the Reactor Building was intended to accommodate large pieces of equipment. The door is bolted internally to the door frame in the shell.

To facilitate reactor pressure vessel removal and dry cask storage, an opening was created in the Reactor Building. The opening has a total length of 58 ft. 8 in. The width of the upper 24 ft. 8 in. of the opening is 16 ft. 9 ¼ in. and the width of the lower 34 ft. is 10 ft. 6 in. The opening is closed by a weather tight, insulated, roll-up, bi-parting door. The opening and door are depicted in Figures 4.6 and 4.7.

Cables and bulkhead conductors from the Turbine Building provide electrical service to the Reactor Building through penetrations in the northwest quadrant of the building shell. The majority of pipe penetrations leave the Reactor Building 1 to 10 ft. below grade level and enter either at the northwest quadrant into the pipe tunnel that runs to the Turbine Building, or on the northeast side into the tunnel connecting the Turbine Building, Reactor Building, stack, and the water treatment and waste gas storage areas.

A 45,000-gal. storage tank in the dome of the Reactor Building supplied water for the emergency core spray system and the building spray system. The storage tank provides a source of water inventory for fuel handling operations and the fuel element storage well.

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

The spent fuel is stored in racks in the bottom of the spent fuel storage well located adjacent to the reactor biological shielding in the Reactor Building. The storage rack system is a two-tier configuration such that each storage location is capable of storing two (2) fuel assemblies, one above the other. Fuel assemblies stored in the lower tier are always accessible (e.g., for periodic inspection) by moving, at most, one other assembly. Each storage rack consists of a welded assembly of fuel storage cells spaced 7 inches on center. A neutron absorbing B_4C/Polymer Composite plate is incorporated between each adjacent fuel storage cell in each orthogonal direction. Horizontal seismic loads are transmitted from the rack structures to the fuel storage well walls at three elevations (the top grid of the upper tier rack section, the top grid of the lower tier rack section and the bottom grid of the lower tier rack section) through adjustable pads attached to the rack structures. The vertical dead-weight and seismic loads are transmitted to the storage well floor by the rack support feet. The fuel storage racks and associated seismic bracing are fabricated from Type 304 stainless steel.

4.2.2 Turbine Building

The general location of the Reactor and Turbine Buildings is shown in Figure 4.3. The Turbine Building contained a major part of the power plant equipment. The turbine-generator was on the main floor. Other equipment was located below the main floor. This equipment included the feedwater heaters, reactor feedwater pumps, air ejector, vacuum pump, full-flow demineralizers, condensate pumps, air compressors, air dryer, oil purifier, service water pumps, component cooling water coolers and pumps, demineralized water system, domestic water heater, turbine oil reservoir, oil tanks and pumps, turbine condenser, unit auxiliary transformer, 2400-volt and 480-volt switchgear, motor control centers, diesel engine-generator sets, emergency storage batteries, inverters and other electrical, pneumatic, mechanical and hydraulic systems and equipment required for a complete power plant. A 30/5-ton capacity, pendant-operated overhead electrical traveling crane spanned the Turbine Building. The crane has access to major equipment items located below the floor through numerous hatches in the main floor. A 40-ton capacity, pendant-operated overhead electric crane spanned the space between turbine building loading dock and Waste Treatment Building.

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

4.2.3 Waste Treatment Building and LSA Storage Building

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

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

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

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

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

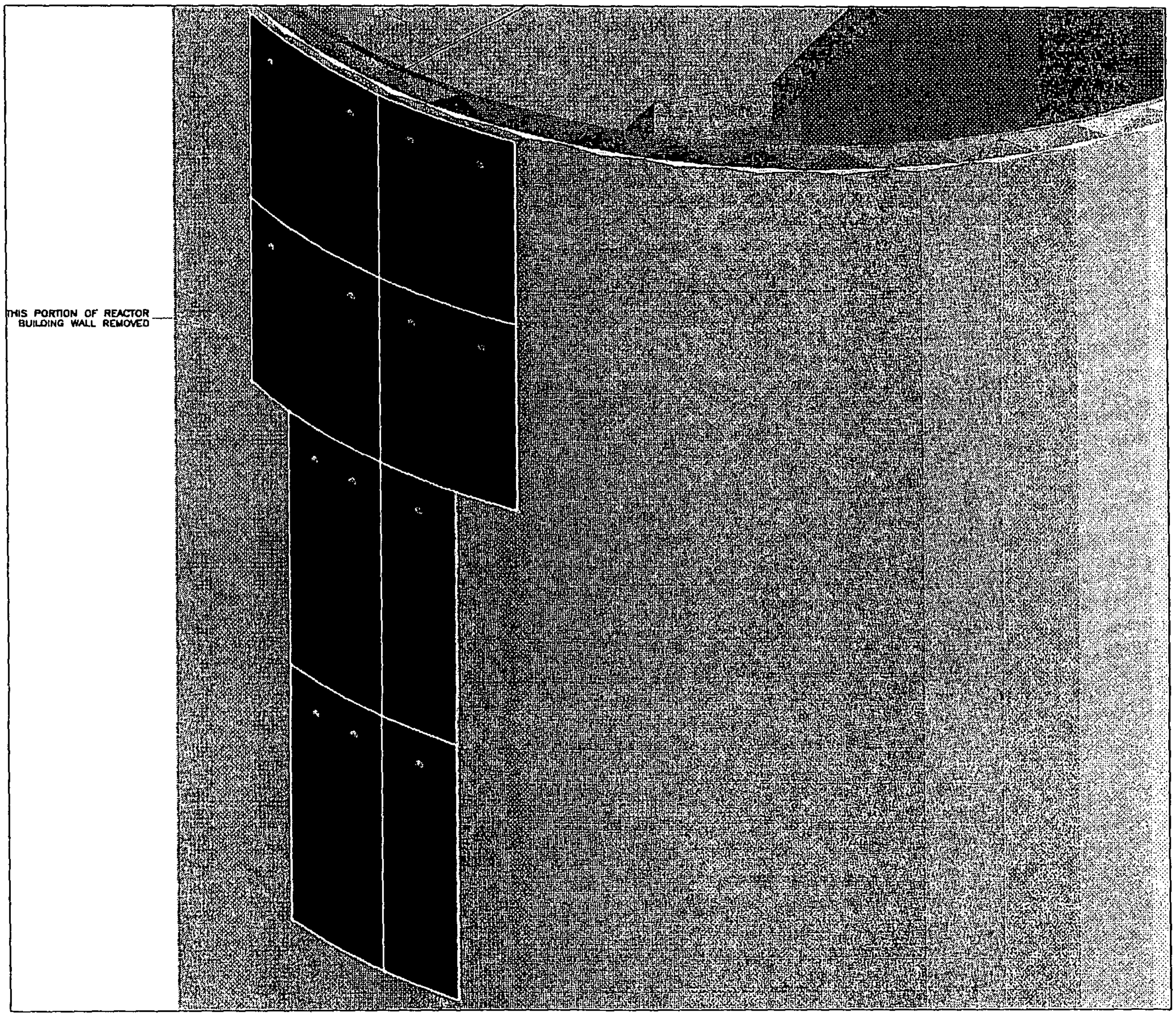
The Waste Treatment Building ventilation is routed through a HEPA filter to the stack plenum. The building is normally maintained at a negative pressure. The general arrangement of the WTB is shown on Figure 4.5.

The LSA Storage Building is southwest of the Turbine Building. It is used to store processed, packaged and sealed low level dry active waste materials, and sealed low level activity components for a period of approximately 5 years. The building has the capacity for 500

DOT17H-55 gallon drums of waste. No liquids are stored in this building. There are no effluent releases from this building during normal use.

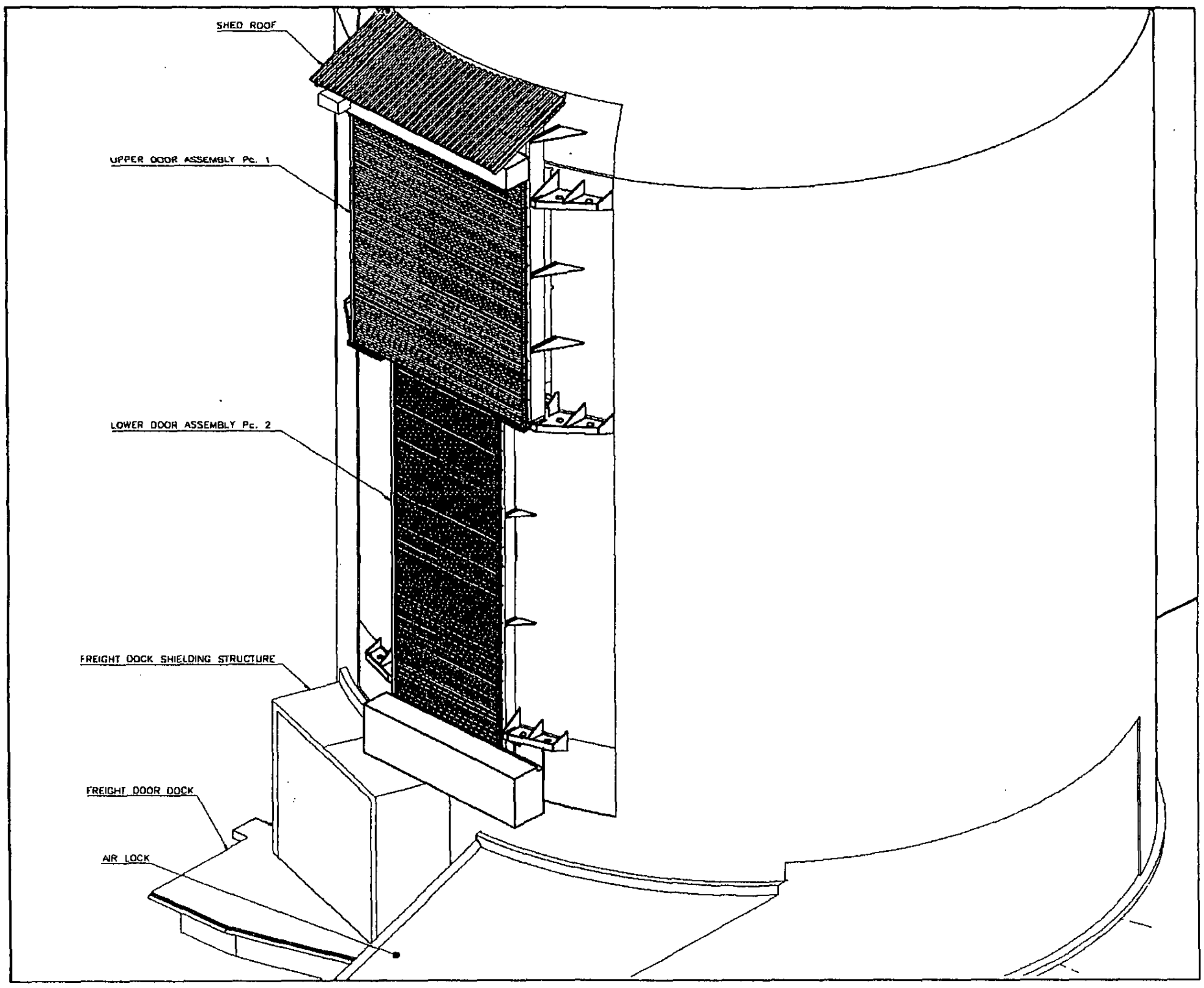
4.2.4 Cribhouse

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



Reactor Building Opening

Figure 4.6



Reactor Building Bi-Parting Door

Figure 4.7

5.2 PLANT SYSTEMS AND THEIR STATUS

5.2.1 Reactor Vessel and Internals

The reactor vessel consisted of a cylindrical shell section with a formed integral hemispherical bottom head and a removable hemispherical top head bolted to a mating flange on the vessel shell to provide for vessel closure. The vessel had 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 was a ferritic steel (ASTM A-302-Gr-B) plate with integrally bonded Type 304L stainless steel cladding. The flanges and large nozzles were ferritic steel (ASTM A-336) forgings. The small nozzles were made of Inconel pipe.

The reactor internals consisted 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 have been removed from the reactor vessel. Startup sources have been disposed of.

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

The reactor pressure vessel was removed from the Reactor Building and shipped to the Barnwell Waste Management Facility in 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 passed upward through the core, and then down through the steam separators to the re-circulating water outlet plenum. The water then flowed to the 16-in. forced circulation pump suction manifold through four 16-in. nozzles and was mixed with reactor feedwater that entered the manifold through four 4-in. connections. From the manifold, the water flowed through 20-in. suction lines to the two 15,000 gpm variable-speed forced-circulation pumps. Hydraulically-operated rotoport valves were installed at the suction and discharge of each pump. The 20-in. pump discharge lines returned the water to the 16-in. forced-circulation pump discharge manifold. From the manifold, the water flowed 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 was 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 was within the biological shield and not accessible, the valves and piping were clad with stainless steel. The piping between the rotoport valves and the pumps was low-alloy steel.

Each forced circulation pump had an auxiliary oil system and a hydraulic coupling oil system. Each auxiliary oil system supplied oil to cool and lubricate the three (1 radial and 2 thrust) pump coupling bearings. Each hydraulic coupling oil system supplied 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 was filled with low density cellular concrete. Four 16-inch forced circulation inlet nozzles and four 16-inch outlet nozzles were cut to allow removal of the reactor pressure vessel. Piping located within the reactor cavity was also cut at the biological shield, segmented into manageable pieces, and disposed of. Pumps and piping in the shielded cubicles remain.

5.4.2 System Radiation Levels

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

The following is a list of the initial survey points, their initial dose rates, and the current survey point dose rates.

Note: All readings are contact dose rates.

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

* Survey Point removed due to dismantlement activities.

Survey Point #	*Survey Point Location*	*Initial Dose Rate (mRem/hr)*	*Current Dose Rate (mRem/hr)*
27	Flash Tank	5	<1
28	Seal Injection Heater	31	*
29	#2 Feedwater Heater Bypass Line	100	*
30	Feedwater Heater Bypass Line	24	*
31	Bottom of Gland Exhaust Condenser	170	*
32	Top of Gland Exhaust Condenser	20	*
33	Condensate into Air Ejector Line	7	*
34	Air Ejector	8	*
35	Low Pressure Turbine Manhole Cover	6	<1
36	End of High Pressure Turbine	2	<1
37	Primary Purification 1A Filter Inlet Line	38	3
38	Primary Purification Pump	140	10
39	Exhaust Ventilation Duct	9	<1
40	Reactor Bldg. Grade Level N Shield Wall	6	<1
41	1A Fuel Element Storage Well Pump	70	1
42	1B Fuel Element Storage Well Pump	80	5
43	FESW Filter Discharge Line	180	10
44	FESW System Cooler	1000	34
45	Hydraulic Valve Actuation System Header	60	2
46	Base of Hydraulic Valve Accumulator	24	<1
47	Wall at Electrical Penetration	30	<1
48	Handrail on NW Nuclear Instrumentation (NI) Platform	100	2
49	Shield Wall on N NI Platform	4	<1
50	Primary Purification to OHST Line	6	<1
51	Above Primary Purification Cooler Inlet Valve	25	5
52	Cold Leg of Reactor High Level Transmitter Line	46	*
53	Seal Injection Reservoir	30	10
54	Reactor Cavity Drain Line	44	4
55	1A Core Spray Pump Discharge Line	10	*
56	Reactor Water Level Sightglass Line	180	*
57	Reactor Water Level Sightglass Line	100	*
58	Reactor Bldg. Mezzanine Level N Shield Wall	4	<1
59	Steam Trap Reactor Bldg. Mezzanine Level NW Wall	23	*
60	Fuel Element Storage Well Line	400	7
61	Fuel Element Storage Well Line	420	7
62	Fuel Element Storage Well Line	60	2

* Survey Point removed due to dismantlement activities.

Survey Point #	*Survey Point Location*	*Initial Dose Rate (mRem/hr)*	*Current Dose Rate (mRem/hr)*
63	Fuel Element Storage Well Skimmer Line	90	3
64	Wall near Fuel Transfer Canal Drain	35	3
65	Relief Valve Platform at Level Transmitter	80	6
66	Shutdwn Condenser	11	*
67	Shutdown Condenser Condensate Line	6	*
68	1B Retention Tank	300	30
69	1A Retention Tank	130	10
70	By Primary Purification Cation Tank	24	2
71	Decay Heat Cooler	25	4
72	Decay Heat Cooler	18	3
73	Decay Heat Cooler Bypass Valve	70	10
74	Decay Heat Pump Suction Line	32	30
75	Handrail at Shutdown Condenser Condensate Valves	28	3
76	Seal Injection DP Transmitter	44	*
77	Top of Upper Control Rod Drive Mechanism	370	65
78	Top of Upper Control Rod Drive Mechanism	200	37
79	Wire mesh screen on N Upper Control Rod Platform	22	2
80	Bottom of Upper Control Rod Drive Mechanism	1000	80
81	Top of Upper Control Rod Drive Mechanism	500	90
82	Bottom of Upper Control Rod Drive Mechanism	800	65
83	Effluent Lines on Upper Control Rod Platform	390	*
84	Sump Pump Discharge Line to Retention Tank	260	16
85	At Forced Circulation Pump Filters	33	3
86	Retention Tank Pump	60	5
87	Under Lower Control Rod Drive Mechanism	246	*
88	Control Rod Drive Hydraulic System Header	190	*
89	Decay Heat Pump	150	15
90	1B Forced Circulation Pump Suction Line	1000	110
91	1B Forced Circulation Pump Suction Line	1100	190
92	1A Forced Circulation Pump Suction Line	500	120
93	1A Forced Circulation Pump Suction Line	600	70
94	1A Forced Circulation Pump Discharge Line	700	110
95	Feedwater Line in Forced Circulation Cubicle	130	22
96	1A Forced Circulation Pump	130	18
97	Handrail at 1A Forced Circ. Pump Suction Line	250	20
98	1A Forced Circulation Pump Discharge Line	800	75
99	1A Forced Circulation Pump Discharge Line	600	70
100	1A Forced Circulation Pump Suction Line	700	50

* Survey Point removed due to dismantlement activities.

6. DECOMMISSIONING PROGRAM

6.1 OBJECTIVES

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

To safely store activated fuel until it can be removed from the site.

To establish a monitoring and surveillance program for comparison to baseline conditions.

To maintain systems required during the SAFSTOR period.

To lay up non-operating systems.

To salvage equipment that is no longer being used.

To handle radioactive waste generated during the SAFSTOR period in accordance with plant procedures and applicable requirements.

To reduce general area radiation levels in the vicinity of equipment operated or maintained during the SAFSTOR period to limit personnel dose to as low as reasonably achievable.

To start decontaminating and dismantling unused systems while minimizing the generation of radioactive waste and personnel dose from this activity.

Maintain qualified and trained staff to fulfill these goals.

6.2 ORGANIZATION AND RESPONSIBILITIES

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

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

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 Maintenance Mechanics are responsible for the completion of all mechanical maintenance tasks. They are responsible for the completion of maintenance requests and surveillance tests of a mechanical nature. They are responsible for the preventive maintenance program established on those systems necessary to maintain the SAFSTOR condition. The Maintenance Mechanics are responsible for overall maintenance on plant equipment which may serve as backups to the required systems or backup supplies to the rest of the Dairyland system.

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 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 complement areas where DPC expertise or staffing is inadequate to perform specific tasks. Contractor employment during the SAFSTOR and DECON periods will continue to be governed by the requirements of the QAPD. Contractors will be selected in each case on a basis of ability, price, past performance, and regulatory requirements.

DPC 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 QAPD, the purchase agreement, and other appropriate regulations.

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

a) The program will be of 12-month duration, and will be repeated each 12 months.

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

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

d) Emergency Plan Training commensurate with duties.

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

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

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

g) The H&S Supervisor will ensure all Journeyman Technicians successfully complete the HP continuing training. Records of satisfactory completion will be maintained by H&S management. The continuing training will cover the following topics.

   (1) Intralaboratory comparisons in radio-chemistry (crosscheck analysis).
   (2) Emergency Plan training.

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

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

   Minutes of these meetings will be taken.

6.4.5 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 current schedule for decommissioning activities at LACBWR is depicted in Figure 6.2. Following final reactor shutdown in April 1987, the transition from operating plant to possession-only facility required numerous administrative changes. Staff level was reduced, license required plans were revised, and operating procedures were curtailed or simplified as conditions and NRC approval allowed. The LACBWR Decommissioning Plan was approved in August 1991, and the facility entered the SAFSTOR mode. License renewal granted at the same time accommodated the proposed SAFSTOR period for a term to expire March 29, 2031.

To make better use of resources during the SAFSTOR period, some incremental decontamination and dismantlement activities were desirable. By Confirmatory Order from the NRC in 1994, changes in the facility meeting 10 CFR 50.59 requirements were permitted and limited gradual dismantlement progressed. As of November 2007, approximately 1.3 million pounds of material related to the removal of unused components or whole systems, completed in over 100 specific approved changes to the facility, have been reprocessed or disposed of as dry active waste. This total does not include reactor vessel and B/C waste disposal.

For decommissioning funding assurance, scheduled DECON had been planned to occur in 2019 and was assumed to be a 7-year project. To date, accumulated decommissioning funds are projected to be sufficient to cover the current decommissioning cost estimate. A possibility exists that DECON may occur earlier. The early start to DECON and a period for License Termination Plan completion are shown on Figure 6.2.

The 2-year Reactor Pressure Vessel Removal (RPV) Project was completed in June 2007 with disposal of the intact RPV at the Barnwell Waste Management Facility (BWMF). Disposal of the RPV was completed at this time prior to the planned closing of BWMF to out-of-compact waste in July 2008. RPV removal was not specifically addressed in the original decommissioning schedule. The removal of this large component, as defined in 10 CFR 50.2, was 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 on August 18, 2005. Section 7.6 describes the RPV Removal Project in greater detail.

The original schedule indicated that during the SAFSTOR period, DPC expected to ship the activated fuel to a federal repository, interim storage facility, or licensed temporary monitored retrievable storage facility. The timing of this action would be dependent on the availability of these facilities and their schedule for receiving activated fuel. In 2007, DPC began efforts to place an Independent Spent Fuel Storage Installation (ISFSI) on-site. As it is titled, the Dry Cask Storage Project has a planned duration of 3 years and is shown on Figure 6.2. An on-site ISFSI is the available option that provides flexibility for license termination of the LACBWR facility. With respect to the federal repository option, a marker for transport of spent fuel to Yucca Mountain has also been added to Figure 6.2 as best available information can provide.

As to another option, DPC is a part of the consortium of utilities that formed the Private Fuel Storage (PFS) Limited Liability Company for the sole purpose of developing a temporary site for the storage of spent nuclear fuel for the industry. The Nuclear Regulatory Commission issued Materials License No. SNM-2513 pursuant to 10 CFR 72, dated February 21, 2006, for the PFS Facility.

At the time of the original Decommissioning Plan in 1987, DPC anticipated the plant would be in SAFSTOR for a 30-50 year period. Since then regulatory guidance has improved and progress in dismantlement at LACBWR has been substantial. B/C waste and RPV disposal have been successfully completed giving momentum to further planning for the final disposition of LACBWR. Development of the Dry Cask Storage Project brings license termination more clearly into focus.

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 Funding

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 Decommissioning Cost Financing

In late 1983, the Dairyland Power Cooperative Board of Directors resolved to provide resources for the final dismantlement of LACBWR. DPC began making deposits to a decommissioning fund in 1984. The Dairyland Power Cooperative Nuclear Decommissioning Trust (DPC—NDT) was established in July 1990 as an external fund outside DPC's administrative control holding fixed income and equity investments. The DPC—NDT, with requisite funding and accumulated earnings, was established to assure adequate funds would be available for the final decommissioning cost of LACBWR.

The cost of DECON was based on the selection of total radiological cleanup as the option to be pursued for LACBWR. At the time of preparation of this plan in 1987, decommissioning cost was based on studies by Nuclear Energy Services, Inc., available generic decommissioning cost guidance, and technology as it existed. In the Safety Evaluation Report dated August 7, 1991, related to the order authorizing decommissioning and approval of the Decommissioning Plan, the NRC found the estimate of $92 million in Year 2010 dollars reasonable for the final dismantling cost of LACBWR.

An improved site-specific decommissioning cost study was performed by Sargent & Lundy (S&L) in 1994 and provides basis for the current cost estimate and funding. The S&L study determined the cost to complete decommissioning to be $83.4 million in Year 1994 dollars with commencement of decommissioning assumed to occur in 2019. A cost study revision completed in July 1998 placed the cost to complete decommissioning at $98.7 million in Year 1998 dollars. A cost study revision, prompted by significant changes in radioactive waste burial costs, as well as lessons learned on decontamination factors and methods, was prepared in November 2000 and 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.

In preparation for removal of the reactor pressure vessel (RPV), cost figures were brought current to $84.6 million in Year 2005 dollars. As of December 2006, DPC—NDT funds were approximately $83.4 million. DPC—NDT funds for B/C waste and RPV removal, approved by the Board of Directors, have been drawn in the amount of $18.2 million. Following B/C waste and RPV disposal a revision to the cost estimate was performed in September 2007 that placed the cost to complete decommissioning at $62.5 million in Year 2007 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 final decommissioning of the LACBWR facility and is prepared to take such 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, Security Force Training and Qualification Plan, Security Control Procedures, Fitness for Duty Program, Unescorted Access Authorization Program, and Behavior Observation Program. The Security Plan for Transportation of LACBWR Hazardous Materials is found in the Process Control Program.

6.11 TESTING AND MAINTENANCE OF SAFSTOR SYSTEMS

Testing and maintenance continues for those systems designated as being required for SAFSTOR. Routine preventive maintenance is performed at specified intervals. Corrective maintenance is performed when identified as necessary. Instrument calibrations and other routine testing continue at specified intervals for equipment required to be operable during SAFSTOR.

The LACBWR Maintenance Rule Program implements requirements of 10 CFR 50.65. The program identifies structures, systems, and components (SSC's) to be monitored under the rule, establishes goals for those SSC's, and provides a process for corrective action implementation for failure of identified SSC's.

6.12 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.

6.12.1 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.

6.12.2 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.

6.12.3 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.

6.12.4 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.

6.13 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.

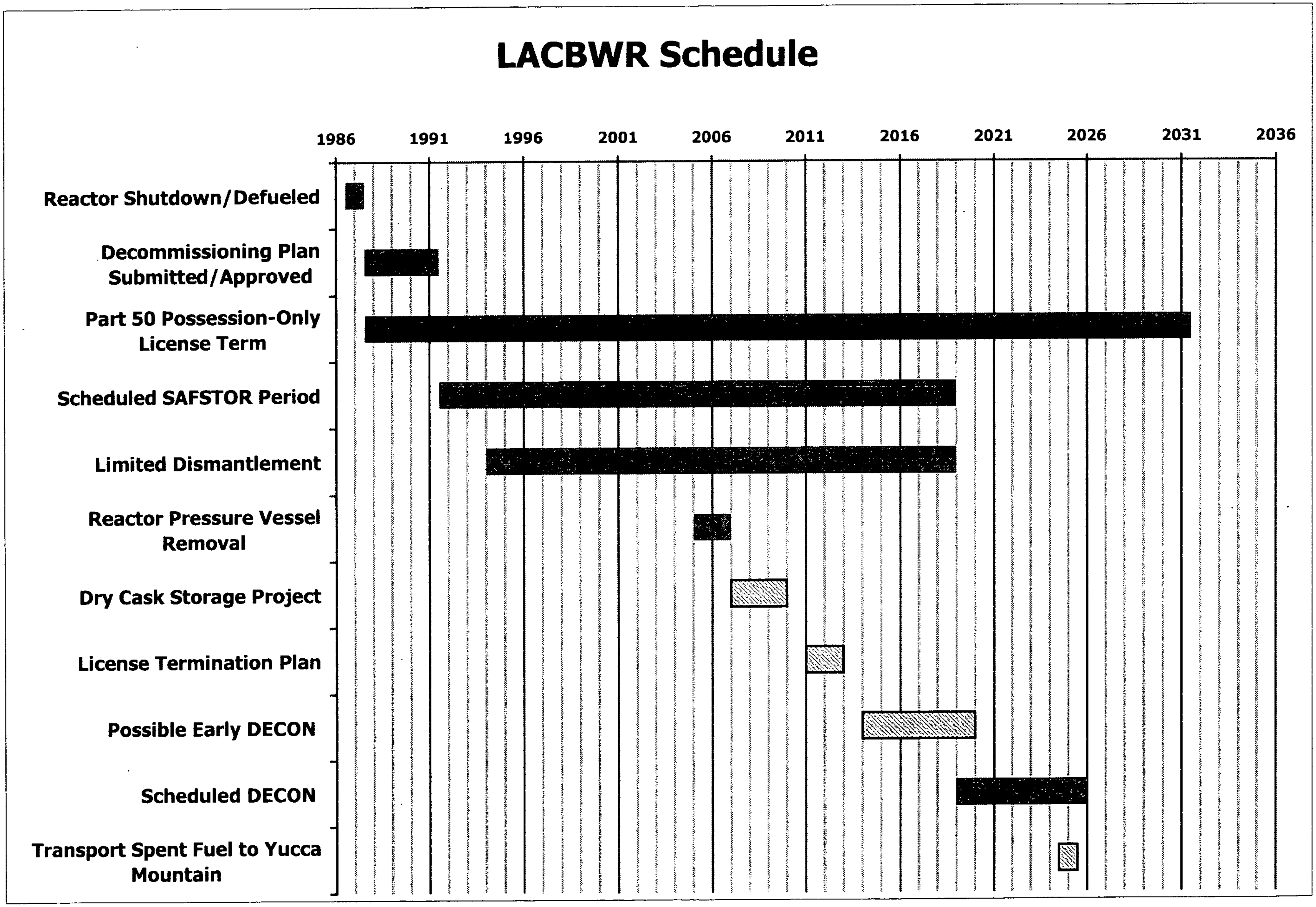

LACBWR Schedule
1986
1991
1996
2001
2006
2011
2016
2021
2026
2031
2036
Reactor Shutdown/Defueled
Decommissioning Plan Submitted/Approved
Part 50 Possession-Only License Term
Scheduled SAFSTOR Period
Limited Dismantlement
Reactor Pressure Vessel Removal
Dry Cask Storage Project
License Termination Plan
Possible Early DECON
Scheduled DECON
Transport Spent Fuel to Yucca Mountain

FIGURE 6.2

7. DECOMMISSIONING ACTIVITIES

7.1 PREPARATION FOR SAFSTOR

The plant was shut down on April 30, 1987. Reactor defueling was completed June 11, 1987. Since the plant shut down, some systems were secured. Additional systems were shut down following determination of lay-up methodology. Others awaited changes to plant Technical Specifications before operational status could be evaluated. Section 5.2 discusses the plant systems and their current status.

In addition to preparation of this Decommissioning Plan, proposed revisions to Technical Specifications, the Security Plan, the Emergency Plan, and the Quality Assurance Program Description were completed. An addendum to the Environmental Report and a preliminary DECON plan were also submitted.

7.2 SAFSTOR MODIFICATIONS

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

The majority involve the Fuel Element Storage Well System (FESW). A redundant FESW level indicator has been added. A second remote manually operated FESW makeup line has been installed, which supplies water from the Overhead Storage Tank. Also, a local direct means of measuring FESW water level has been installed.

The air activity monitoring system has been replaced with new equipment. The gas activity monitors have been recalibrated to a Kr-85 equivalent. Kr-85 will be the predominant gaseous isotope during the SAFSTOR period.

7.3 SIGNIFICANT SAFSTOR LICENSING ACTIONS

DPC's authority to operate LACBWR under Provisional Operating License DPR-45, pursuant to 10 CFR Part 50, was terminated by License Amendment No. 56, dated August 4, 1987, and a possess but not operate status was granted. The Decommissioning Plan was submitted December 1987 with a chosen decommissioning alternative of SAFSTOR. License Amendment No. 63, dated August 18, 1988, amended the operating license to Possession-Only License DPR-45 with a term to expire March 29, 2003.

The NRC directed the licensee to decommission the facility in its Decommissioning Order of August 7, 1991. License Amendment No. 66, issued with the Decommissioning Order and also of the same date, provided evaluation and approval of the proposed Decommissioning Plan, proposed SAFSTOR Technical Specifications, and license renewal to accommodate the proposed SAFSTOR period for a term to expire March 29, 2031.

The Decommissioning Order was modified September 15, 1994, by Confirmatory Order to allow the licensee to make changes in the facility or procedures as described in the Safety Analysis Report, and to conduct tests or experiments not described in the Safety Analysis Report, without prior NRC approval, if a plant-specific safety and environmental review procedure containing similar requirements as specified in 10 CFR 50.59 was applied.

The Initial Site Characterization Survey for SAFSTOR was completed and published October 1995 and is attached as revised to this Decommissioning Plan.

License Amendment No. 69, containing the SAFSTOR Technical Specifications, was issued April 11, 1997. This amendment revised the body of the license and the Appendix A Technical Specifications. The changes to the license and Technical Specifications were structured to reflect the permanently defueled and shutdown status of the plant. These changes deleted all requirements for emergency electrical power systems and maintenance of containment integrity.

The SAFSTOR Decommissioning Plan is considered the post-shutdown decommissioning activities report (PSDAR). The PSDAR public meeting was held on May 13, 1998.

Review of and revisions to this Decommissioning Plan, the Security Plan, the Emergency Plan, the Quality Assurance Program Description, the Offsite Dose Calculation Manual, and other material, continue at intervals as required.

7.4 AREA AND SYSTEM DECONTAMINATION

The decontamination program during the SAFSTOR period will be a continuation of routine decontamination work performed at LACBWR. Plant areas and component outer surfaces will be decontaminated to reduce the requirements for protective equipment use and to reduce the potential for the translocation of radioactive material. Decontamination methods that are used are dependent upon a number of variables, such as surface texture, material type, contamination levels, and the tenacity with which the radioactive material clings to the contaminated surfaces.

Surface areas are primarily decontaminated using hand wiping, wet mopping, and wet vacuuming techniques. Detergents and other mild chemicals may be used with any of these techniques. The residual water cleaning solutions are collected by floor drains and processed through the liquid waste system. Most areas are routinely decontaminated to levels below 2000 dpm/ft^2 (about 500 dpm/100 cm^2). Many areas are maintained below the Lower Limit of Detection (LLD). Efforts will be made to maintain all accessible areas in the plant as free of surface contamination as is reasonably achievable.

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

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

7.5 REMOVAL OF UNUSED EQUIPMENT DURING SAFSTOR

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

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

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

Removal of plant equipment will be performed only after review. A 10 CFR 50.59 review will be conducted prior to dismantling any system.

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

7.6 REACTOR PRESSURE VESSEL REMOVAL

DPC entered contract agreement July 2005 with Duratek, Inc. (later to become Energy Solutions, LLC) for removal and disposal of the intact Reactor Pressure Vessel (RPV) at the Barnwell Waste Management Facility (BWMF) in South Carolina. Major subcontractors included Bigge Power Constructors, ARES Corp., Bluegrass Concrete Cutting, Inc., Pacific International Grout Co., and Patent Construction Systems. Engineering and project development progressed through 2005 and 2006. The RPV, forced circulation loop piping, and pumps were filled with low density cellular concrete (LDCC) in March 2006. The LDCC fixed in place components and contamination internal to the RPV. Site mobilization and major project activity commenced September 2006.

The opening in the Reactor Building (RB) required removal of one polar crane runway support column. Restoration of the polar crane to full capacity by installation of an alternate support plate and components was completed to allow use of the polar crane in support of project activities. Three upper cavity shield plugs, 15' diameter, 15" thick, weighing 30 tons each, were staged, cut into 18 pieces, and removed to clear obstruction to the project. Portions of the concrete floor at elevation 701' were cut and removed to clear travel path obstruction and provide access to the octagonal biological shield (bio-shield) structure. Two floor and beam shoring supports were installed.

Core drilling was performed in areas of the RB wall and the bio-shield openings for rigging and cutting wire access. Cores were drilled to allow precise diamond wire saw cuts that created manageable sized blocks. An opening in the steel and concrete exterior RB wall was completed, then closed by installation of a weather-tight, insulated, roll-up, bi-parting door in November 2006. The RB opening (described in Section 4.2.1) and bi-parting door are depicted in Figures 4.6 and 4.7.

A 10'-6" opening was made in the 4' to 6' thick concrete bio-shield from elevation 701' down to elevation 667'. From access in the cavity, RPV nozzles and appurtenances were cut from the RPV to near bottom dead center and to a critical diameter of 119" by March 2007. Temporary lifting device erection and installation began as nozzle cutting was being completed.

The grout-filled RPV weighing 370,000 lbs. was disconnected from its support, lifted 20', translated outside the RB, and placed upright into a staged steel cylindrical package. The package with RPV was filled with concrete, seal welded, down-ended, and heavy-hauled to an on-site rail siding. The RPV package weighing 624,500 lbs. was loaded onto a special transport rail car, and shipped with final burial at the BWMF completed June 6, 2007.

7.6.1 <u>Temporary Lifting Device</u>

A Temporary Lifting Device/Gantry Rail System (TLD) was erected and installed inside and outside the RB. The TLD system consisted of a temporary runway structure and rolling trolley which incorporated hydraulic strand jacks for lifting the RPV. All TLD equipment was removed following RPV removal with the exception of two rocker bearing assemblies installed on the bio-shield at elevation 701' and two bearing assemblies mounted at the RB wall opening.

(1) <u>Runway Structure Inside the RB</u>

- Inside girders were installed in 37' spans on each side of the 701' floor cut opening. The girders were a fabricated plate steel box design with runway rails installed on the top flange. Each inside box girder was supported at the south end by a rocker bearing assembly attached by bolts to the bio-shield structure, and at the north end by a cantilever bearing assembly attached to the end of the 74' outside girder sitting on the RB wall. The inside 37' girders did not bear directly on the 701' floor.

- The box girders used to support the TLD rail were from Bigge inventory and had existing bearing mounting plate extensions from the ends of the girder box structure. These extensions were not used in the RPV removal configuration; however, the southeast girder bearing extension did overhang the FESW by nine inches. The TLD rail, which was installed on top of the girder, did not overhang the FESW; therefore, the TLD rail-stop was located north of the FESW north wall. TLD travel for RPV removal was from the south to north direction.

- Each box girder had a runway rail travel stop limit mounted on the south end to prevent trolley movement beyond the end of the girders toward the FESW. The south end of the east girder was located at the edge of the FESW north wall. The south end of the west girder was approximately 3' from the FESW north wall. The girders ran away from the FESW with the load path centerline at 16.5 degrees east of the RB centerline north. Loaded trolley travel was away from the FESW.

- Girder support and restraint:

  - At the south end of the runway girders, vertical support, lateral restraint, and longitudinal restraint were provided for each girder by the rocker bearing assembly at the bio-shield/FESW wall.

  - Vertical support and lateral restraint were provided for each girder by the bearing assembly at the RB wall.

- The runway structure design inside the RB met NUREG-0612 criteria.

(2) <u>Runway Structure Outside the RB</u>

- The box girder sections outside the RB were 74' span. Each girder was supported by a bearing assembly located at the RB wall and by the bent structure at the north end of the runway.

- Each box girder had a runway rail travel stop limit mounted on the north end to prevent the trolley from rolling off the north end of the runway.

- The outside box girders extended approximately 2'-6" into the RB beyond the RB wall mounted bearing assembly. At this location the girders had a stepped web which accommodated a rocker bearing for the inside 37' girders to bear upon.

- Girder support and restraint:

  - Longitudinal restraint for the outside runway structure was provided by the previously described bio-shield/FESW mounted rocker bearings.

  - Lateral restraint for the north end of the outside 74' girders and support bent was provided by the counter weight structure located west of the support bent. Lateral restraint for the south end of the outside girders was provided by the bearing assemblies mounted on the RB wall.

- The runway structure design outside the RB was not required to meet NUREG-0612 criteria, as NUREG-0612 pertains to lifts and equipment inside buildings where spent fuel is stored.

(3) Trolley, Jacks, Rigging, and Fixtures

- The trolley was a moveable platform with four two-wheeled bogie end trucks (8 total double flanged wheels) designed to run on the box girder rails. Two of the trucks had electric mechanical drives. Each drive consisted of a gearbox, motor, and brake. There were two driven/braked wheels in the 8 wheel set. The brake was automatically set when the momentary directional motion switch was released to the neutral position.
- The trolley had two travel speeds; 1.62 feet per minute (FPM) and 6.80 FPM. Both travel speeds were very slow. At the slower speed it would have taken over an hour to traverse the runway from south to north.
- Two hydraulic strand jack hoisting systems were mounted on top of a trolley platform. The strand jack systems were independent from each other and were specially fabricated to meet the specifications for the LACBWR RPV lift and transport. Hoisting speed was 0.5 FPM.
- The strand jacks were comprised of 36 strands per jack; failure of any given strand would not result in loss of control of the suspended load. Failure of over 75% of the strands would have had to occur before the remaining strands could not carry the load.
- Two separate electrical sources were used to power the two strand jack power packs and one trolley drive system through three dedicated load disconnect switches. The strand jack system was designed such that the load would remain secured at the height lifted upon loss of power or hydraulic pressure.
- Emergency power disconnect switches, located separately from the operator station for each strand jack hydraulic power pack and trolley drive, were provided to remotely stop the systems in the event of a hydraulic spray leak or other off normal condition.
- The trolley assembly was designed to meet NUREG-0612 criteria.

(4) Load Testing

The TLD was constructed of components within the Bigge equipment inventory along with new fabricated assemblies. Prior to TLD use for the RPV lift, a load test of 110% of the load lifted outside the RB (service load 639,000 lbs/test load 703,000 lbs) was conducted. Since a load test of 150% of the load lifted inside the RB (service load 380,000 lbs/test load 570,000 lbs) was less than the outside load test weight, the inside load test was not performed. The percent increases above static weight or service load were consistent with NQA-1 and ANSI N14.6.

The custom built RPV attachment/handling fixture used inside the RB was load tested in accordance with ANSI N14.6-1993, Section 7, "Special lifting devices for critical loads." Section 7.3.1(a) required the test load to be three times (3x) the weight the fixture would support. The handling fixture load test was documented for record.

7.6.2 NUREG-0612 Compliance

RPV removal project activities that occurred inside the RB were performed in compliance with NUREG-0612 due to the close proximity of the RPV lift to the stored spent fuel. Additional information relative to NUREG-0612, and TLD compliance with other applicable codes and standards, was provided in Bigge Document No. 2150-D1. Compliance to NUREG-0612, Section 5.1, was met by implementing the following measures for each of the criteria:

5.1(1) The project implemented detailed operator training, handling system design, load handling instructions, and conducted equipment inspection to ensure reliable operation of the handling system.

5.1(2) The safe load travel path was included in procedures and operator training. The engineered load path was such that the RPV was not carried over or near irradiated fuel or important to safety (ITS) SSCs.

5.1(3) Mechanical stops were provided to prevent movement of heavy loads over irradiated fuel or in proximity to ITS equipment.

Guidance in NUREG-0612, Section 5.1 allows for certain deficiencies if alternate compensatory measures are credited; however, no deficiencies in defense-in-depth criteria were noted. Additional measures were included that could be given credit for. The items providing defense-in-depth were: increased hoisting system reliability by providing an increased factor of safety, increased inspections, and significant spent fuel decay. The lift supervisor also held Certified Crane Operator (CCO), as designated by the National Commission for the Certification of Crane Operators.

Facility Change 30-06-08, "RPV Supports Disconnection, Lift and Transit," documented compliance with the seven criteria in NUREG-0612, Section 5.1.1 for overhead handling systems that handle heavy loads in the area of the spent fuel pool. The criteria fully addressed were: (1) safe load paths; (2) procedures; (3) crane operators training and qualification; (4) codes and standards for special lifting devices; (5) codes and standards for lifting devices not specifically designed; (6) codes and standards for crane inspection, testing and maintenance; and (7) codes and standards for crane design.

NUREG-0612, Section 5.1.4 discusses boiling water reactors, but does not address the RPV as a potential heavy load. Section 5.1.4 states that in addition to meeting the general guidelines of Section 5.1.1, either the lifting devices should meet the guidelines in Section 5.1.6, or the effects of a heavy load drop should be analyzed. TLD design met the requirements of Section 5.1.6. The runway and trolley structural steel were designed to twice (2x) the design safety factor to meet the requirements of AISC and for SSE seismic conditions with the lifted load to the design criteria of Bigge Document No. 2150-D1. Jacking components were designed to twice (2x) normal design safety factors of ANSI B30.1, and below the hook lifting devices were designed to twice (2x) normal design safety factors in accordance with ANSI N14.6-1993, Section 7. The in-place structure was designed for normal and seismic load cases with the lifted load to meet plant design basis acceptance limits.

TLD design met NUREG-0612, Section 5.1.6(3) requirements in that all interfacing lift points such as lifting lugs or trunnions had a design safety factor of ten times (10x) the maximum static plus dynamic load.

TLD design met the defense-in-depth approach. Although there were no deficiencies, certain additional measures could be given credit. The TLD was used solely for lifting and removal of the RPV. Compliance to NUREG-0612 was in the context that the TLD was not planned as new permanent plant equipment or replacing the existing polar crane, but was used as a Temporary Lifting Device to lift the RPV.

7.6.3 50.59 Evaluations

RPV Removal Project work was performed under nine major Facility Changes (FCs) for which 50.59 Evaluations were conducted and are listed below:

- FC 30-06-05, LACBWR Reactor Pressure Vessel Grouting
- FC 37-06-31, Biological Shield Cutting and Removal
- FC 37-06-35, 50/5-Ton Polar Crane Runway Restoration
- FC 37-06-32, Reactor Building Modification
- FC 37-06-34, Reactor Building Restoration Activities
- FC 30-06-06, RPV Bottom Head Nozzle and Appurtenances Removal
- FC 30-06-07, Reactor Pressure Vessel Nozzle Removal
- FC 37-06-33, TLD/Gantry System Install, Test and Disassembly
- FC 30-06-08, RPV Supports Disconnection, Lift and Transit

These 50.59 Evaluations examined RPV Removal Project activities under eight criteria and concluded that performance of each activity did not:

(1) Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the Decommissioning Plan.

(2) Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety previously evaluated in the Decommissioning Plan.

(3) Result in more than a minimal increase in the consequences of an accident previously evaluated in the Decommissioning Plan.

(4) Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the Decommissioning Plan.

(5) Create a possibility for an accident of a different type than any previously evaluated in the Decommissioning Plan.

(6) Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the Decommissioning Plan.

(7) Result in a design basis limit for fission product barrier, as described in the Decommissioning Plan, being exceeded or altered.

(8) Result in a departure from a method of evaluation described in the Decommissioning Plan used in establishing the design bases or in the safety analyses.

Calculations and analyses in support of the conclusions of these 50.59 Evaluations are listed in Section 7.6.4. RPV Removal Project work was performed under twenty-three additional FCs for which 50.59 Screens were adequate in determining that the proposed activities created no new failure modes or other adverse effects.

7.6.4 References

(1) Bigge Document 2150-D1, "Engineering Design Basis, Engineering, Rigging and Onsite Transport Services, Phase 2 – Reactor Pressure Vessel Removal Project, La Crosse Boiling Water Reactor Nuclear Plant," April 26, 2007, Rev. 2.

(2) Bigge Calculation No. 2150-C10, "Strand Jack Trolley Adequacy," Rev. 1.

(3) Bigge Calculation No. 2150-C30, "TLD Runway Structure," Rev. 2.

(4) Bigge Calculation No. 2150-C50A, "RPV Lift Lug and Miscellaneous Hook Rigging," Rev. 0.

(5) Bigge Calculation No. 2150-C50B, "RPV Head Adequacy," Rev. 0.

(6) Bigge Calculation No. 2150-CTLD Seismic, "TLD Runway & Trolley Structure Seismic," Rev. 0.

(7) ARES Calculation No. 0526301.11-S-001, "Regeneration of LACBWR 1982 Containment Building Model for Seismic and Structural Analysis," Rev. 0.

(8) ARES Calculation No. 0526301.11-S-002, "Seismic Analysis of Modified LACBWR Containment Building with SAP2000," Rev. 1.

(9) ARES Calculation No. 0526301.11-S-003, "Structural Analysis of LACBWR Modified Containment Building Outer Shield Wall to Support Crane Girder Loads During RPV Removal," Rev. 1.

(10) ARES Calculation No. 0526301.11-S-005, "Structural Reinforcing of the Containment Building Outer Shield Wall Opening to Maintain Polar Crane Capacity," Rev. 0.

(11) ARES Calculation No. 0526301.11-S-006, "Structural Analysis for Shoring of Floor at El. 701' Due to Concrete Cutting Inside the Reactor Building," Rev. 0.

(12) ARES Calculation No. 0526301.11-S-007, "Structural Integrity Analysis of Spent Fuel Storage Well and Racks Inside the Reactor Building," Rev. 1.

(13) ARES Calculation No. 0526301.11-S-008, "Seismic Analysis of LACBWR's Temporary Lifting Device (TLD) Structure for Removing RPV from the Reactor Building," Rev. 0.

(14) ARES Calculation No. 0526301.12-S-001, "Structural Analysis of Support Reinforcement for Bi-Parting Door at Containment Building Opening," Rev. 0.

(15) ARES Report No. 0526301.11-002, "Phase 2, LACBWR Reactor Pressure Vessel Removal Structural Analysis and Design Criteria," Rev. 0.

7.7 B/C WASTE REMOVAL

Included in the scope of work during the RPV project was removal of irradiated hardware and other B/C wastes. Processing of waste stored in the FESW began in April 2006. This waste consisted of 73 irradiated zircaloy fuel shrouds, 24 irradiated stainless steel fuel shrouds, 10 irradiated control rod blades, and 2 antimony-beryllium startup sources. A control rod extension shear, control rod blade crimper, and hydraulic crusher shear were used to process components into cylindrical waste liners. Startup sources were placed in liners without processing. The Duratek CNS 3-55 Shipping Cask was used in the transport of two liners of irradiated hardware waste and disposal was completed at the BWMF in July 2006. Other B/C waste included resins, filters, and waste barrel contents that were collected in three liners and shipped for disposal in June 2007.

7.8 ENVIRONMENTAL IMPACT

Review of post-operating license stage environmental impacts was documented in a supplement to the Environmental Report for LACBWR dated December 1987. LACBWR dismantlement and decommissioning activities have resulted in no significant environmental impact not previously evaluated in the NRC's Environmental Assessment in support of the August 7, 1991, Decommissioning Order or the Final Environmental Statement (FES) related to operation of LACBWR, dated April 21, 1980 (NUREG-0191).

The environmental impact of all completed or planned limited dismantlement activities is SMALL as determined by the "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities (GEIS)," NUREG-0586, Supplement 1, November 2002. The environmental impact of ISFSI construction and operation at the LACBWR site is an activity not within the scope of the GEIS and will be addressed in the licensing process for the activity.

Site-specific potential environmental impacts not determined in the GEIS are:

- Offsite land use activities
- Aquatic ecology as to activities beyond the operational area
- Terrestrial ecology as to activities beyond the operational area
- Threatened and endangered species
- Socioeconomic
- Environmental justice

The License Termination Plan (LTP) for LACBWR will detail final dismantlement activities, processes for demolition of structures, site remediation, survey of residual contamination, and determination of site end-use. A final supplement to the Environment Report in support of the LTP will address all environmental impacts of the license termination stage.

8.2.2 Application of ALARA

(1) To obtain the goal of ALARA, the Total Effective Dose Equivalent (TEDE) to be received during a specific job and the total allowable for the year should be balanced to the extent possible.

(2) The occupational dose received by an individual shall be considered with respect to his/her yearly internal, and external accumulation. The individual's TEDE dose should be balanced with the TEDE dose received by other members of his/her department to the extent possible.

(3) An SWP ALARA review may be conducted if the following thresholds are expected to be exceeded:

(a) Between 100 and 500 milliRem total collective deep dose equivalent (DDE) for performing a job.

(b) Potential intake greater than 50 DAC-HRS for an individual and respiratory devices are not planned to be used. An ALARA review form used for this application should be governed by total Person Rem and DAC-HR estimates, based upon current surveys and job-time estimates, and total Person Rem for past similar jobs based upon an SWP dose accountability file.

(4) An SWP ALARA review shall be conducted if the following thresholds are expected to be exceeded:

(a) Greater than 500 milliRem collective DDE for performing a job.

(b) Potential intake greater than 100 DAC-HRS for an individual and respiratory devices are not planned to be used.

(5) If a job is expected to require greater than 1.0 Person Rem, a more intensive ORC ALARA Review shall be required. This may include the use of special procedures.

(6) Documentation of ALARA engineering work and cost benefits shall be maintained in files.

(7) HP Management will conduct ALARA reviews of Actual versus Projected (goal) exposures. Person Rem exposures will be reviewed regularly with the Plant Manager. Included should be a review of the effectiveness of specific steps that were taken to reduce radiation exposure (ALARA Engineering).

8.2.3 Radiation Exposure Limits

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

(2) Occupational Dose Limit Guideline
LACBWR will provide a guideline for the control of occupational dose to individual adults to the following annual limits. Any individual exceedance of these limits will require approval by the Plant Manager.

a) Total effective dose equivalent (TEDE) 2.5 Rem.

b) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 25 Rem.

c) An eye dose equivalent of 7.5 Rem

d) A shallow-dose equivalent of 25 Rem to the skin or to each of the extremities.

e) The dose received by an embryo/fetus during the entire pregnancy due to the occupational exposure of a declared pregnant worker shall not exceed 0.25 Rem (250 mRem).

(3) Occupational Dose Limit
LACBWR shall control the occupational dose to individual adults to the following annual limits. Any individual exceedance of these limits will require approval by the ORC.

a) Total effective dose equivalent (TEDE) less than 4 Rem.

b) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 40 Rem.

c) An eye dose equivalent of 12 Rem.

d) A shallow-dose equivalent of 40 Rem to the skin or to each of the extremities.

e) The dose received by an embryo/fetus during the entire pregnancy due to the occupational exposure of a declared pregnant worker shall not exceed 0.4 Rem (400 mRem).

(4) The NRC establishes the following occupational dose limits to individual adults, except for Planned Special Exposures authorized under 10CFR20.1206, to the following annual limits:

a) Total Effective Dose Equivalent (TEDE) 5 Rem.

b) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye being equal to 50 Rem.

c) An eye dose equivalent of 15 Rem.

d) A shallow-dose equivalent of 50 Rem to the skin or to each of the extremities.

e) The dose received by an embryo/fetus during the entire pregnancy due to the occupational exposure of a declared pregnant worker should not exceed 0.50 Rem (500 mRem).

(5) The Health and Safety Department shall be contacted as soon as possible when entry into an area of known or suspected high airborne radioactivity has or will occur. The Health and Safety Department shall provide air sampling and recommend protective measures for entry to such areas. Engineering measures that can be taken to reduce or eliminate the airborne contamination areas are the recommended methods to reduce/eliminate internal deposition. Respiratory protection will be used only after a review to ensure an excessive increase in the external dose received, due to the use of respirators, will not occur. Suspected exposure to airborne contamination for inhalation/ingestion or waterborne concentrations of radioactive materials shall be investigated by lung deposition counting and/or urinalysis.

(6) An adult worker may be authorized to receive doses in addition to and accounted for separately from the doses received under the normal occupational limits. These are known as Planned Special Exposures and will be used only under unusual/emergency situations. Planned Special Exposures will be authorized only in accordance with 10 CFR 20.1206.

8.3 RADIATION PROTECTION PROGRAM

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

8.3.1 Personnel Monitoring

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

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

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

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

Long-term visitors expecting to receive a radiation exposure of 50 mRem will be issued badges and dosimeters and will be monitored in the same manner as the regular plant personnel.

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

Badge records received from the badge processor will be evaluated and maintained. Periodic quality testing of badges and pocket dosimeters will be conducted.

Bioassays will be performed in accordance with the requirements of 10 CFR 20.1204 and in conformance to the recommendations of Regulatory Guide 8.26, "Application of Bioassay for Fission and Activation Products," and Regulatory Guide 8.32, "Criteria for Establishing a Tritium Bioassay Program."

The LACBWR internal lung deposition counter will be used to detect any internal lung contamination for:

a) All new employees who will routinely work with radioactive material.

b) Any individual suspected of having received any internal lung deposition.

c) Upon termination of any employee who worked with radioactive material.

If it is determined that any employee has a significant internal lung deposition of any isotope, the individual may be required to submit a urine and/or fecal specimen.

All personnel leaving a restricted area will be required to conduct a personnel contamination survey using the contamination detection instrument provided at the exit.

8.3.2 Respiratory Protection Program

A respiratory protection program will be maintained during the SAFSTOR period.

The Health and Safety Supervisor is responsible for the Respiratory Program at LACBWR. The Health and Safety Supervisor or designated alternate will evaluate the total job hazard, recommend engineering controls if appropriate, specify respiratory protection if control cannot be otherwise obtained and forbid the use of respirators if conditions warrant. The Health and Safety Department is responsible for the selection, care, and maintenance of all respiratory protection equipment that falls under the scope of the respiratory protection program.

The acceptable manner for limiting the internal exposure of personnel is to control radioactivity concentration in the air breathing zones. Whenever possible, this will be accomplished by the application of engineering control measures such as containment, decontamination, special ventilation equipment and design. The use of personal respiratory protective equipment as a primary control is undesirable and is acceptable only on a non-routine basis or in an emergency situation.

Equipment such as hoods, blowers, and filtered exhaust systems will be used to provide controls for routine operations and, whenever possible, for non-routine operations. In some cases, such controls may be inadequate or impractical and the use of protective breathing apparatus will be approved on a short-term basis.

The periods of time for which respirators may be worn continuously, and the overall time of uses, should be kept to a minimum. The wearer shall leave the area for relief from respirator use in case of equipment malfunction, undue physical or psychological discomfort, or any other condition that, in the opinion of the user, his supervisor or the Health and Safety Department, might cause significant reduction in the protection afforded the user.

Respiratory protection equipment will be issued to individuals only after documentation has been received that shows that the person has satisfactorily completed:

a) medical exam,

b) respiratory protection training, and

c) respiratory fit test (does not apply to in-line supplied air hoods and Self-Contained Breathing Apparatus).

8.3.3 Protective Clothing

Personnel working in contaminated areas of LACBWR are provided with protective clothing to minimize the potential for personnel contamination. Routine entry into a contaminated area will require a minimum protective clothing requirement of:

a) coveralls
b) head covering
c) gloves
d) shoe coverings

Specific jobs may require additional protective clothing. These additional requirements will be determined by the Health and Safety Department and will be listed on the Special Work Permit for the job.

During the SAFSTOR period, the laundry facility will remain operational to ensure an adequate supply of clean protective clothing.

8.3.4 Access Control

To limit radiation exposures, personnel access is controlled in areas where such exposure is possible. This control consists of a system of physical barriers, warning signs and signals.

A Special Work Permit (SWP) will be issued as authorization for personnel to perform work of a non-routine nature in a specific area which involves unusual hazards. SWP's will be used to inform personnel of these hazards and the safeguards/protective measures which need to be taken during the work to ensure their
well being.

8.3.5 Postings

Postings shall be in accordance with the requirements of 10 CFR 20, Subpart J, as applicable.

8.4 RADIATION MONITORING

A program for routine surveys and monitoring will be continued during the SAFSTOR period at LACBWR. This program will continue to assure all personnel are aware of the possible hazards involved before entering a potential radiation area or a potentially contaminated area. This will be done to ensure that the potential hazards are adequately defined, that adequate controls are instituted so that radiation exposure to personnel working in radiation areas or working with radioactive materials is minimized, and that each person carries out his work in a radiologically safe manner.

Survey data records will be maintained to assist in the evaluation of the radiological conditions and trends at LACBWR during SAFSTOR activities.

The radiological monitoring program will include the following surveys:

a) airborne activity surveys
b) dose rate surveys
c) contamination surveys
d) liquid activity surveys
e) environmental surveys

8.4.1 Airborne Radioactivity Surveys

In addition to using the fixed location or mobile air monitors, particulate airborne activity shall also be determined as needed by drawing a sufficient quantity of air through a filter paper. The samples shall be counted for beta-gamma activity in gas-flow proportional detector and scaler equipment. Alpha activity of a sample shall be determined by means of a windowless gas-flow proportional detector and a scaler when alpha radioactivity is suspected of being present. Samples are analyzed for specific isotopic concentrations, by the use of a gamma analyzer. Particulate samples of the stack releases will be obtained and analyzed weekly to determine release rates.

Non-routine air samples to establish protection requirements for maintenance activities or to verify airborne radioactivity conditions during work activities are obtained and analyzed when routine samples are not sufficient for monitoring plant conditions.

8.4.2 Radiation Surveys

Radiation surveys are conducted for the following purposes:

a) Measure and document radiation and contamination levels in areas of interest.

b) Identify trends in radiation and contamination levels, particularly during work in progress.

c) Determine appropriate protective measures for personnel working in restricted areas.

d) Provide information so that workers can maintain their doses ALARA.

e) Identify locations and situations where special dosimetry is required.

In addition to the measurements made by the fixed-location area radiation monitors, the measurement of external dose-rates shall be accomplished by portable survey instruments. The operation of the survey instruments shall be in accordance with the operating instructions outlined in each particular instrument manual or by procedure. Instruments covering high, intermediate, and low ranges shall be available on site.

Surveys will be conducted by the Health and Safety Department to determine general area dose rates. They will also monitor areas to locate any radiological hot spots. Surveys will be performed on a routine basis established by procedures.

Special radiation surveys of particular items or areas are performed on an "as needed" basis. Examples of special radiation surveys are the removal of equipment or materials from a restricted area, leak testing of sealed radioactive sources, or the shipment or receipt of radioactive material packages.

8.4.3 Contamination Surveys

Contamination surveys will be conducted routinely by the Health and Safety Department as established by procedure to determine area contamination levels.

Special contamination surveys of particular items or areas are performed on an "as needed" basis. Examples of special contamination surveys are the removal of equipment or materials from a restricted area, leak testing of sealed radioactive sources, or the shipment or receipt of radioactive material packages.

A dry filter paper or cloth disc will be wiped over approximately one square foot (12"x12" square or 12'-long S-shaped) of the surface being monitored. Swipes will be counted for beta-gamma activity in a gas-flow proportional detector or with a 2 π GM probe or equivalent in fixed geometry sample holder as necessary. Alpha activity of a swipe will be determined by means of a windowless gas-flow proportional detector and a scaler or equivalent, when alpha radioactivity is suspected of being present.

8.4.4 Liquid Activity Surveys

Samples of water containing radioactivity are collected and analyzed on a routine basis. Spent fuel pool water is analyzed to detect indications of degradation of the fuel stored in the pool. Samples of liquid radioactive wastes and processed wastes are analyzed to ensure levels of radioactivity are below the levels permitted for release. Samples are analyzed by Health and Safety Department personnel in accordance with established procedures.

8.4.5 Environmental Surveys

Environmental samples will be taken within the surrounding areas of the plant. These samples will be analyzed to determine any effects plant effluent releases may have on the environment. This program will be conducted as per the ODCM.

8.5 RADIATION PROTECTION EQUIPMENT AND INSTRUMENTATION

A variety of equipment and instruments are used as part of the radiation protection program. Equipment and instrumentation are selected to perform a particular function. Sensitivity, ease of operation and maintenance, and reliability are factors that are considered in the selection of a particular instrument. As the technology of radiation detection instrumentation improves, new instruments are obtained to more accurately measure radioactivity and ensure an effective radiation protection program.

This equipment can be broken down into several specific groups each with its own dedicated functions. These groups are:

a) Portable Instruments
b) Installed Instruments
c) Personnel Monitoring Instruments
d) Counting Room Instruments

This equipment will be used, checked and calibrated by trained personnel according to in-plant procedures.

8.5.1 Portable Instruments

There will be sufficient types and quantities of portable instruments to provide adequate beta, gamma, and alpha surveys at LACBWR. This equipment will have the ability to detect these types of radiation over the potential ranges that will be present during SAFSTOR. Portable dose rate instruments will be source checked prior to use, and they will be calibrated semiannually.

8.5.2 Installed Instrumentation

There will be sufficient types and quantities of installed instrumentation to provide continuous in-plant and effluent release monitoring. This will assure the safe reliable monitoring of both area dose rates and airborne activity concentration throughout the area. These instruments will be response tested monthly and calibrated once every 18 months.

8.5.3 Personnel Monitoring Instrumentation

Friskers and personnel instrumentation monitors will be provided throughout the plant to provide personnel contamination monitoring. These monitors will be of the type and sensitivities necessary to minimize the spread of in-plant contamination and prevent the introduction of contamination to outside areas. This equipment will be checked daily during normal workdays and calibrated semiannually.

8.5.4 Counting Room Instrumentation

Laboratory equipment will be available to perform gross alpha and beta analyses and gamma isotopic analyses of samples collected in the plant. There will also be equipment available in a low background area to provide adequate analysis of environmental samples. A quality control program will be in effect for this equipment to ensure the accurate and proper operation of the equipment. Gross alpha/beta counters will be calibrated annually. The HPGe detectors will be calibrated every two years.'

8.6 RADIOACTIVE WASTE HANDLING AND DISPOSAL

Radioactive waste generated at LACBWR during the SAFSTOR period will primarily consist of the following:

a) Resin
b) Dry active waste (DAW)
c) Dismantlement (Metallic)

Radioactive waste generation will be maintained as low as possible to minimize the volume of material requiring reprocessing and disposal.

8.6.1 Resin

Spent resin will be transferred to the spent resin receiving tank where it will be held until there is a sufficient quantity available for shipment to an approved processing facility. The resin will be transferred to an approved shipping container where it will be dewatered and made ready for shipment.

8.6.2 Dry Active Waste (DAW)

Any material used within the restricted area will be considered radioactive and will be disposed of as DAW, unless it can be demonstrated to be within established releasable limits. The generation of this material will be maintained as low as possible to reduce the total waste volume generated onsite. The material generated will be placed into approved shipping containers.

8.6.3 Dismantlement (Metallic)

During the SAFSTOR period, LACBWR employees will pursue limited dismantlement of the facility. This project will generate metallic wastes from system removal. This metallic waste will be placed in approved shipping containers and sent to an approved reprocessor.

Disposal of all radioactive waste will be in accordance with all pertaining guidelines.

8.7 RECORDS

Records generated in the performance of the radiation protection program will be maintained as required to provide the necessary documentation of the program and in accordance with the QAPD. These records will be maintained in a designated storage area.

8.8 INDUSTRIAL HEALTH AND SAFETY

LACBWR will continue to participate in Dairyland Power Cooperative's industrial safety program as prescribed by the DPC Safety Department. These programs will include:

a) Accident prevention
b) Hazardous waste management and control
c) Asbestos control
d) Hearing conservation

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 2007 only 27.5% of the Kr-85 activity remains - 118.6 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

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

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 E-2 Ci/sec x 2.3 E-4 sec/m^3 = 1.38 E-5 Ci/m^3

Immersion Dose Conversion at 500m E

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

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

Whole Body Dose at 500m E

$$1839 \frac{\text{mRem/hr}}{\text{Ci/m}^3} \text{ x } 1.38 \text{ E-5 Ci/m}^3 \text{ x } 2 \text{ hr} = 0.05 \text{ mRem}$$ (as of 10/07 = 0.01 mRem)

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

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

Skin Dose at 500m E

$$1.53\ \mathrm{E5}\ \frac{\mathrm{mRem/hr}}{\mathrm{Ci/m}^3}\ x\ 1.38\ E\text{-}5\ \mathrm{Ci/m}^3\ x\ 2\,\mathrm{hr} = 4.2\ \mathrm{mRem}\ (\text{as of } 10/07 = 1.2\ \mathrm{mRem})$$

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 E-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
10/07 = 0.13 mRem	10/07 = 11.1 mRem

Ground Level Release at Emergency Planning Zone Boundary

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

1.02 E-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
10/07 = 0.60 mRem	10/07 = 51.4 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. 2007 only 27.5% of the Kr-85 activity remains – 9,834 Curies.)

Elevated Release

Whole Body Dose at 500m E	Skin Dose at 500m E
10/87 = 4.2 mRem	10/87 = 350 mRem
10/07 = 1.2 mRem	10/07 = 96.3 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
10/07 = 11.1 mRem	10/07 = 0.92 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
10/07 = 51.2 mRem	10/07 = 4.3 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.

LAC-TR-138

LACBWR

INITIAL

SITE CHARACTERIZATION SURVEY

FOR SAFSTOR

By:

Larry Nelson
Health and Safety Supervisor

October 1995

Revised: November 2007

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

2.0 OPERATING EVENTS WHICH COULD AFFECT PLANT CLEANUP

(1) Failed Fuel

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

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

(2) Fuel Element Storage Well Leakage

The stainless steel liner in the Fuel Element Storage Well (FESW) has had a history of leakage. From the date of initial service until 1980, the leakage increased from approximately 2 gallons per hour (gph) to just over 14 gph. In 1980, epoxy was injected behind the liner and leakage was reduced to approximately 2 gph. In 1993, the FESW pump seals were discovered to be defective and were replaced which reduced the leak rate to approximately 1 gph. FESW leakage has stabilized over the years to an average of approximately 21 gallons per day. This leakage is contained within the steel shell of the Reactor Building.

(3) Release of Contaminated Water To The Controlled Area, July 2, 1982 at 0630

The failure to close the resin inlet valve in the resin addition line to the number one full flow demineralizer following the addition of resins subsequently caused the release of water to the Turbine Building Floor through the gasket on a bulls-eye sight glass in that line. Approximately 75 gallons of contaminated water could not be accounted for in the waste water storage tanks. Approximately 20 gallons of this water is estimated to have entered the ground in the radiological controlled area outside the west turbine hall door and the turbine hall truck bay door. Contaminated ground was removed over a 3 sq. ft area by the west turbine hall door and 2 sq. yard area by the truck bay door. It was placed in waste storage barrels and later sent to burial at Barnwell, S.C.

ATTACHMENT 1

SPENT FUEL RADIOACTIVITY INVENTORY

Decay-Corrected to October 2007

Radionuclide	*Half Life (Years)*	*Activity (Curies)*	*Radionuclide*	*Half Life (Years)*	*(Curies)*
Ce-144	7.801 E-1	6.1 E-2	Sr-90	2.770 E + 1	6.99E+5
Cs-137	3.014 E+1	1.06 E+6	Pu-241	1.429 E+1	4.39E+5
Ru-106	1.008 E+0	1.87	Fe-55	2.700 E+0	3.26E+3
Cs-134	2.070 E+0	435	Ni-59	8.000 E+4	287
Kr-85	1.072 E+1	3.23E+4	Tc-99	2.120 E+5	276
Co-60	5.270 E+0	4.73E+3	Sb-125	2.760 E+0	1.89
Pm-147	2.620 E+0	219	Eu-155	4.960 E+0	10.6
Ni-63	1.000 E+2	3.09E+4	U-234	2.440 E+5	63.7
Am-241	4.329 E+2	1.43E+4	Am-243	7.380 E+3	63
Pu-238	8.774 E+1	1.08 E+4	Cd-113m	1.359 E+1	6.49
Pu-239	2.410 E+4	8.83E+3	Nb-94	2.000 E+4	15.9
Pu-240	6.550 E+3	7.15E+3	Cs-135	3.000 E+6	14.0
Eu-154	8.750 E+0	838	U-238	4.470 E+9	12.2
Cm-244	1.812 E+1	1.69E+3	Pu-242	3.760 E+5	8.58
H-3	1.226 E+1	180	U-236	2.340 E+7	6.32
Eu-152	1.360 E+1	186	Sn-121m	7.600 E+1	3.71
Am-242m	1.505 E+2	447	Np-237	2.140 E+6	2.19
			U-235	7.040 E+8	1.89
			Sm-151	9.316 E+1	1.3
			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 = 2.31 E6 Curies

ATTACHMENT 2

CORE INTERNAL/RX COMPONENT RADIONUCLIDE INVENTORY – OCTOBER 2007

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)	REACTOR VESSEL WAS PROCESSED, PACKAGED AND DISPOSED OF IN 2007				
Control Rods (29)					
Core Vertical Posts (52)					
Core Lateral Support Structure					
Steam Separators (16)					
Thermal Shield					
Pressure Vessel					
Core Support Structure					
Horizontal Grid Bars (7)					
Incore Monitor Guide Tubes					
Total					
In FESW					
Fuel Shrouds (24 SS)	All "In FESW" components listed were processed, packaged, and shipped for disposal in 2006.				
Fuel Shrouds (73 Zr)					
Control Rods (10)					
Start-up Sources (2)					

ATTACHMENT 3

PLANT SYSTEMS INTERNAL RADIONUCLIDE INVENTORY - OCTOBER 2007

Plant System	Nuclide Activity, in µCi				System Total µCi Content
	Fe-55	Alpha	Co-60	Cs-137	
CB Ventilation	10	--	118	108	236
Offgas - upstream of filters	*SYSTEM*	*REMOVED*			
Offgas - downstream of filters	*SYSTEM*	*REMOVED*			
TB drains	106	40	1,258	3,170	4,574
CB drains	236	3	2,812	1,521	4,572
TB Waste Water	22	7	266	76	371
CB Waste Water	1,304	79	15,540	1,458	18,381
Main Steam	1,614	290	19,240		21,144
Turbine	6	2	69	127	204
Primary Purification	552	12	6,586		7,150
Emergency Core Spray	*SYSTEM*	*REMOVED*			
Overhead Storage Tank	81	34	962	494	1,571
Seal Inject	10	4	118	35	167

ATTACHMENT 3

PLANT SYSTEMS INTERNAL RADIONUCLIDE INVENTORY – OCTOBER 2007 - (cont'd)

Plant System	Nuclide Activity, in µCi			System Total µCi Content
	Fe-55	Alpha	Co-60	
Decay Heat	621	490	7,400	8,511
Boron Inject	*SYSTEM*	*REMOVED*		
Reactor Coolant PASS	*SYSTEM*	*REMOVED*		
Alternate Core Spray	124	94	1,480	1,698
Shutdown Condenser	*SYSTEM*	*REMOVED*		
Control Rod Drive Effluent	931	720	11,100	12,751
Forced Circulation	9,310	7,000	111,001	127,311
Reactor Vessel and Internals	*SYSTEM*	*REMOVED*		
Condensate after beds & Feedwater	*SYSTEM*	*REMOVED*		
Condensate to beds	*SYSTEM*	*REMOVED*		

ATTACHMENT 3

PLANT SYSTEMS INTERNAL RADIONUCLIDE INVENTORY – OCTOBER 2007 - (cont'd)

Plant System	Nuclide Activity, in μCi				System Total μCi Content
	Fe-55	Alpha	Co-60	Cs-137	
Fuel Element Storage Well System	5,276	390	62,901		68,567
Fuel Element Storage Well - all but floor	8	5	96	2,916	3,025
Fuel Element Storage Well floor	161,395	7,600	1,924,017	25,990	2,119,002
Resin lines	807	100	9,620		10,527
Main Condenser	68,282	8,500	814,007		890,789