WILLIAM L. BERG
President and CEO

COR-FILE: D-2

DAIRYLAND POWER
COOPERATIVE

November 29, 2006

In reply, please refer to LAC-13948

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:tco
Enclosures

cc: Kris Banovac
NRC Project Mgr.

NMSS01

2006 LACBWR Decommissioning Plan Review

Cover Page	Update revision date.
Page 5-3	Section 5.2.1, Reactor Vessel and Internals: The first paragraph of "System Status" is revised to two sentences stating, ***"All fuel assemblies have been removed from the reactor vessel. Startup sources have been disposed of."*** The second paragraph, second sentence is revised to state, *"The reactor vessel with head installed, internals intact, and 29 control rods in place **has been filled** with low density cellular concrete."* Changes are to update information of the LACBWR Reactor Pressure Vessel Removal Project and B&C waste removal.
Page 5-4	Section 5.2.2, Forced Circulation System: The first sentence of the second paragraph of "System Status" is revised to update information of the LACBWR Reactor Pressure Vessel Removal Project by stating, *"All 16-inch and 20-inch forced circulation system piping **has been filled** with low density cellular concrete as part of the Reactor Pressure Vessel Removal Project undertaken in August 2005.* Change updates information with completion of reactor pressure vessel grouting.
Page 5-10	Section 5.2.8, Alternate Core Spray System: Second sentence of "System Status" is revised by the following, *"The 6-inch supply line to the reactor pressure vessel head **has been** removed as interference to removal of the reactor pressure vessel."* Purpose of change is to update information of the LACBWR Reactor Pressure Vessel Removal Project.
Page 5-13	Section 5.2.11, Fuel Element Storage Well System: Fourth paragraph, first sentence is revised stating, *"Spent fuel elements are stored in two-tiered racks in the Fuel Element Storage Well until **removed to cask storage.**"* Under "System Status" the last three sentences are deleted that state, *"Also stored in the well are 10 control rods, 2 antimony-beryllium startup sources, 24 stainless steel fuel element shroud cans, and 73 zirconium alloy fuel element shroud cans. These components will be removed, packaged, and disposed of. Removal of irradiated hardware and other B&C waste has been included in the scope of work during the Reactor Pressure Vessel Removal Project."* Changes correct terminology and update information following B&C waste removal.
Page 5-20	Section 5.2.18, Overhead Storage Tank: Third paragraph, first and second sentences are revised to reflect change in system capability due to Reactor Vessel Removal Project stating, *"The Overhead Storage Tank (OHST) **served** as a reservoir for water used to flood the Fuel Element Storage Well, **reactor vessel,** and upper vessel cavity during **refueling.** The OHST **acted** as a receiver for rejecting **refueling** water using the Primary Purification System."* The following statement is added to "System Status" to be consistent with Section 5.2.11, ***"A 4-inch line from the OHST is used to flood the well or pump water back to the OHST using FESW System pumps, valves, and piping."***

Page 5-38	Section 5.2.34.1, Reactor Building Atmosphere PASS System Description: The paragraph is revised following completion of an approved facility change to state, *"The Reactor Building Atmosphere Post-Accident Sampling System consists of a vacuum pump that takes suction from the Reactor Building atmosphere at the 714' elevation.* ***The sample is drawn through a bypass line or to a remote sample cylinder and discharged back to the Reactor Building at the 671' elevation."***
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 2006 values.
Page 5-44	Section 5.6, Sources: Paragraph is revised to state, *"As authorized by the facility license, sealed sources for radiation monitoring equipment calibration will continue to be possessed **and** used. **Additionally, sources will be used as** authorized without restriction to chemical or physical form for sample analysis,* ***instrument calibration, and as associated with radioactive apparatus and*** *components."* Purpose of change is to update information following B&C waste removal; mention of reactor related sources has been deleted.
Page 6-9	Section 6.4.5, Other Decommissioning Training: Section is renumbered to ***6.4.4*** to correct information as written in original 1987 D-Plan.
Page 6-10	Section 6.4.6, Training Program Administration and Records: Section is renumbered to ***6.4.5*** to correct information as written in original 1987 D-Plan.
Page 6-11	Section 6.6, Schedule: Section continues from previous page. Second paragraph, second sentence is revised to reflect PFS licensing status by stating, *"The Nuclear Regulatory Commission* ***issued Materials License No. SNM-2513 pursuant to 10 CFR 72, dated February 21, 2006, for the PFS Facility."***
Page 6-19	Section 6.10, Security during SAFSTOR and/or Decommissioning: In the second paragraph, mention of the following recently installed programs has been added: ***"Fitness for Duty Program, Unescorted Access Authorization Program, Behavior Observation Program."*** *"Guard Force"* is corrected at ***"Security*** *Force Training and Qualification Plan."*
Page 7-2	Section 7.3.1, Significant SAFSTOR Licensing Actions: Fifth paragraph, first sentence is revised to state, *"License Amendment No. 69, containing the* ***SAFSTOR*** *Technical Specifications, was issued April 11, 1997."* Change is to correct information. A minor change has since occurred with the issuance of License Amendment No. 70, the current Technical Specifications, which accommodated B&C waste removal operations in the Fuel Element Storage Well.
Page 8-4	Section 8.3.1, Personnel Monitoring: Section continues from previous page. In discussion that includes reference to *"internal deposition,"* clarification is added

by changes to *"internal lung deposition."* Change better describes equipment capability and methodology. A single change from the masculine pronoun *"he"* is made to the genderless form ***"the individual"*** in the third paragraph of the section.

Page 9-2 Through Page 9-3	Section 9.2, Spent Fuel Handling Accident: The curie content remaining as of *October 2005* and calculated values for Whole Body Dose and Skin Dose as of *October 2005* are updated to ***October 2006.***
Page 9-4	Section 9.3, Shipping Cask or Heavy Load Drop into FESW: The curie content remaining as of *October 2005* and calculated values for Whole Body Dose and Skin Dose as of *October 2005* are updated to ***October 2006.***
Page 9-5 Through Page 9-6	Section 9.5, FESW Pipe Break: Accident analysis is revised by deleting discussion of increase in radiation levels from control rods in FESW being exposed with loss of water level. Control rods have been processed and disposed of as part of B&C waste removal. Accident analysis is also revised to reflect that decay heat from the stored fuel has decreased with passage of time. Review of testing performed and documented, ***"indicates boiling in the FESW is not expected to occur,"*** rather than, *"boiling would not commence for more than one (1) day."* Additional information of this testing is added and the accident analysis is updated to reflect current conditions and assumptions. The final paragraph of the Section 9.5 accident analysis on page 9-6 is revised in two paragraphs to state:

"There would be no immediate urgency to restore the level. No release of contamination is associated with this event. Active FESW cooling would be lost during this accident, ***but a test conducted during July 1993 with normal water level in the FESW indicates that*** *considerable time is available to take action.* ***This test, as documented in LACBWR Technical Report, LAC-TR-137, showed that with all cooling and coolant circulation to the pool isolated, FESW water temperature increased from 80°F to only 114°F in 15.5 days. This test was terminated at 114°F to limit increasing radioactivity in the pool water. Extrapolation of the data indicated the temperature would have stabilized at approximately 150°F.*** *Due to the* ***smaller*** *water volume to act as the heat sink in the FESW pipe break accident the initial heat up rate of the FESW water would be approximately twice as great as that during the 1993 test. The heat removal rate from the FESW at a given temperature will be reduced somewhat since the wetted wall area is reduced by approximately 42%. The heat removal at a given temperature, by evaporation of the FESW coolant and condensation on the FESW cover and walls, will be essentially unchanged. Since heat removal rate increases rapidly as the temperature in the FESW increases, engineering judgment indicates the temperature in the FESW will stabilize at a temperature somewhat above 150°F, but boiling is not expected to occur. The total heat source in the FESW is only about 12.2 kW."***

"As with the loss of FESW cooling event, if water is added to the FESW, any consequences of water heat up can be delayed or prevented. Water can be added from the Demineralized Water System or the Overhead Storage Tank."

Page 9-7 Through Page 9-8	Pages are reissued due to content shift from previous page with no other changes.

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

Cover Page	Update revision date.
Page 14	Table title is missing from page and is added.
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 2006,*** replacing pages that had been decay-corrected to *October 2005.*
Page 25	Curie content is deleted at *"In FESW"* section of table and is replaced by note stating, ***"All "In FESW" components listed were processed, packaged, and shipped for disposal in 2006."***

LA CROSSE BOILING WATER REACTOR

(LACBWR)

DECOMMISSIONING PLAN

Revised
November 2006

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

5. PLANT STATUS – (cont'd)

5.2 PLANT SYSTEMS AND THEIR STATUS

5.2.1 Reactor Vessel and Internals

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

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

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

System Status

All fuel assemblies have been removed from the reactor vessel. Startup sources have been disposed of.

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

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

5.2.2 Forced Circulation System

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

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

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

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

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

System Status

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

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

5.2.8 Alternate Core Spray System

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

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

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

System Status

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

5. PLANT STATUS – (cont'd)

5.2.11 Fuel Element Storage Well System

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

It is completely lined with Type 316 stainless steel. The walls are 16-gauge sheet and the bottom a 3/8-inch plate. All joints are full penetration welds. Vertical and horizontal expansion joints in the storage well allow for thermal expansion. A three-section aluminum cover, with two viewing windows per section, has been manufactured to cover the pool.

Design values for the storage well are given below:

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

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

Storage well cooling is accomplished by drawing water through a 6-inch penetration at elevation 679 feet, or a 4-inch line at elevation 679 feet 11 inches, and pumping it through the fuel storage well cooler and returning it to the well, with either of two storage well pumps. The return line enters the top of the storage well and extends down to discharge at elevation 695 feet. The bottom inlet line ends at the biological shield wall and is sealed with a welded plug.

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

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

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

System Status

The Fuel Element Storage Well contains 333 irradiated fuel elements and will remain in operation as part of the SAFSTOR Program as long as wet fuel storage or wet fuel handling is necessary.

5. PLANT STATUS – (cont'd)

5.2.18 Overhead Storage Tank

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

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

The Overhead Storage Tank (OHST) served as a reservoir for water used to flood the Fuel Element Storage Well reactor vessel and upper vessel cavity during refueling. The OHST acted as a receiver for rejecting refueling water using the Primary Purification System. The OHST supplied the water for the Emergency Core Spray System and Reactor Building Spray System, and was a backup source for the Seal Injection System.

System Status

The Overhead Storage Tank remains in use, primarily for a source of makeup water to the Fuel Element Storage Well. A 4-inch line from the OHST is used to flood the well or pump water back to the OHST using FESW System pumps, valves, and piping.

5.2.34.1 Reactor Building Atmosphere PASS System Description

The Reactor Building Atmosphere Post-Accident Sampling System consists of a vacuum pump that takes suction from the Reactor Building atmosphere at the 714' elevation. The sample is drawn through a bypass line or to a remote sample cylinder and discharged back to the Reactor Building at the 671' elevation.

5.2.34.2 Stack Gas PASS System Description

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

5.2.34.3 Reactor Coolant PASS System Description

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

System Status

The Stack Gas PASS System is maintained in continuous operation. The Reactor Coolant PASS System has been removed. The Reactor Building Atmosphere PASS System is retained in place.

5.2.35 Containment Integrity Systems

With the plant in the SAFSTOR condition, there is no longer a postulated accident that would result in containment pressurization or that takes credit for Containment integrity.

System Status

Containment integrity systems are not required to be operable.

5. PLANT STATUS – (cont'd)

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	7
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	9
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.

5. PLANT STATUS – (cont'd)

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	12
39	Exhaust Ventilation Duct	9	<1
40	Reactor Bldg. Grade Level N Shield Wall	6	<1
41	1A Fuel Element Storage Well Pump	70	4
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	6
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.

5. PLANT STATUS – (cont'd)

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	36
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
087	Under Lower Control Rod Drive Mechanism	246	*
88	Control Rod Drive Hydraulic System Header	190	20
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.

5.5 PLANT PERSONNEL DOSE ESTIMATE

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

5.6 SOURCES

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

5.7 RADIATION MONITORING INSTRUMENTATION

Radiation monitoring instrumentation for the LACBWR consists of fixed plant surveillance equipment, portable survey meters, laboratory-type counting instrumentation, and personnel monitoring equipment.

The Radiation Monitoring System performs the following functions:

(1) Provides a permanent record of radioactivity levels of plant effluents.

(2) Provides alarms and automatic valve closures to prevent excessive radioactive releases to environment.

(3) Provides warning of leakage of radioactive gas, liquid, or particulate matter within the plant.

(4) Provides continuous radiation surveillance in normally accessible plant areas.

(5) Provides portable instrumentation for use in conducting radiation surveys.

(6) Provides instrumentation for personnel and material contamination surveillance, including that necessary for control of egress from restricted areas.

(7) Provides pocket dosimeters and necessary charging and readout equipment for personnel radiation exposure control and estimates.

(f) Emergency Training

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

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

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

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

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

During the SAFSTOR period, it is not expected that movements of spent reactor fuel will be made, except for special tests or inspections to monitor the fuel in storage. At some time during the SAFSTOR period, fuel handling may be performed to transfer the spent fuel assemblies to the Department of Energy (DOE) or other entity.

6.4.4 Other Decommissioning Training

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

6.4.5 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 tentative decommissioning schedule is shown in Figure 6-2. As can be seen, DPC received a possession-only license in August 1987. The LACBWR Decommissioning Plan was approved in August 1991, and the facility entered the SAFSTOR mode.

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

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

During the SAFSTOR period, DPC expects to ship the activated fuel to a federal repository, interim storage facility, or licensed temporary monitored retrievable storage facility. The timing of this action will be dependent on the availability of these facilities and their schedule for receiving activated fuel. A modification to the Decommissioning Plan will then be submitted to describe the change in plant status and associated activities.

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

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

6.7 SAFSTOR FUNDING AND DECOMMISSIONING COST FINANCING

DPC is currently assuming a 30-50 year SAFSTOR period. For cost estimating purposes, however, it was assumed that dismantlement commences as soon as possible, which would be shortly after the fuel is sent to a federal repository. The year 2011 was chosen as the earliest possible for DECON to commence. SAFSTOR and DECON costs are funded separately. SAFSTOR funding accommodates management of LACBWR spent fuel and provides assurance of continued funding through all modes of fuel storage prior to acceptance by the DOE. Mandated decommissioning funds will be available during the DECON period.

6.7.1 SAFSTOR

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

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

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

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 RECORDS

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

Any records which are generated for the safe and effective decommissioning of LACBWR will be placed in a file explicitly designated as the decommissioning file.

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

- Records of spills or spread of radioactive contamination, if residual contamination remains after cleanup.
- Records of contamination remaining in inaccessible areas.
- Plans for decontamination (including processing and disposal of wastes generated).
- Base line surveys performed in and around the LACBWR facility.
- Analysis and evaluations of total radioactivity concentrations at the LACBWR facility.
- Any other records or documents, which would be needed to facilitate decontamination and dismantlement of the LACBWR facility and are not controlled by other means.

7.3 ACTIVITIES DURING SAFSTOR PERIOD

7.3.1 Significant SAFSTOR Licensing Actions

The licensee's authority to operate Facility License No. DPR-45, pursuant to 10 CFR Part 50, was terminated by license Amendment No. 56, dated August 4, 1987, and a possession-only status was granted. The decommissioning alternative of SAFSTOR was chosen.

The NRC directed the licensee to decommission the facility in its Decommissioning Order of August 7, 1991. 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.

License Amendment No. 66, issued with the Decommissioning Order and also dated August 7, 1991, provided evaluation and approval of the proposed Decommissioning Plan, proposed SAFSTOR Technical Specifications, and license renewal to accommodate the proposed SAFSTOR period until March 29, 2031.

The Initial Site Characterization Survey for SAFSTOR was completed and published October 1995 and is attached 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.

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

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 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 2006 only 29.3% of the Kr-85 activity remains - 126 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.

<u>Elevated Release</u>

Average Kr-85 Release Rate

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

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

Kr-85 average concentration at 500m E

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

Immersion Dose Conversion at 500m E

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

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

Whole Body Dose at 500m E

$$1839 \frac{\text{mRem/hr}}{\text{Ci/m}^3} \times 1.38 \text{ E-5 Ci/m}^3 \times 2 \text{ hr} = 0.05 \text{ mRem} \quad (\text{as of } 10/06 = 0.02 \text{ mRem})$$

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

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

Skin Dose at 500m E

$$1.53\ \mathrm{E}5\ \frac{\mathrm{mRem/hr}}{\mathrm{Ci/m}^3}\ x\ 1.38\,E-5\ \mathrm{Ci/m}^3\ x\ 2\,\mathrm{hr} = 4.2\ \mathrm{mRem}\ (\text{as of } 10/06 = 1.2\ \mathrm{mRem})$$

Ground Level Release at EAB

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

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

Whole Body Dose at 338m	Skin Dose at 339m
10/87 = 0.49 mRem	10/87 = 40.4 mRem
10/06 = 0.14 mRem	10/06 = 11.8 mRem

Ground Level Release at Emergency Planning Zone Boundary

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

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

Whole Body Dose at 100m E	Skin Dose at 100m E
10/87 = 2.25 mRem	10/87 = 187 mRem
10/06 = 0.70 mRem	10/06 = 54.8 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. 2006 only 29.3% of the Kr-85 activity remains – 10,478 Curies.)

Elevated Release

Whole Body Dose at 500m E	Skin Dose at 500m E
10/87 = 4.2 mRem	10/87 = 350 mRem
10/06 = 1.2 mRem	10/06 = 102.6 mRem

Ground Level Release at EAB

Whole Body Dose at 338m	Skin Dose at 338m
10/87 = 40.2 mRem	10/87 = 3.34 Rem
10/06 = 11.8 mRem	10/06 = 1.0 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
10/06 = 54.5 mRem	10/06 = 4.6 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.

9.4 LOSS OF FESW COOLING

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

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

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

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

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

After the final discharge of fuel to the FESW, a conservative calculation of the FESW heatup rate was performed using the estimated decay heat source in the spent fuel on January 1, 1988. This calculation indicated that coolant boiling could occur approximately 5 days after the loss of cooling.

In July 1993, a test was conducted to determine the actual heat-up rate of the FESW with all cooling and coolant circulation to the pool isolated. This test, as documented in LACBWR Technical Report, LAC-TR-137, showed that the pool temperature increased from 80°F to only 114°F in 15.5 days. The test was terminated at 114°F to limit increasing radioactivity in the pool water, but extrapolation of the data indicates the temperature would stabilize at approximately 150°F.

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

9.5 FESW PIPE BREAK

This accident postulates a break in the FESW system piping, other than in the pump discharge piping between the redundant check valves and the pool liner. A load analysis was performed on this approximately 20 feet of piping. It was concluded that all stresses are within ASME Code allowable. (Reference 1 calls this line the spent fuel pool drain line.) The series check valves were added during the 1980 FESW reracking. In November 1999, the FESW return line was rerouted to enter the top of the storage well and extends down to discharge at elevation 695 ft. The bottom inlet line now ends at the biological shield wall and is sealed with a welded plug.

If the postulated break occurs, the lowest the FESW could drain is approximately 679 ft. At this level all spent fuel will remain covered. The operator would be alerted to this accident by receipt of the FESW Level Lo/High alarm. Any makeup water added may run out the break, depending on the size of the break. In the vicinity of most of the FESW piping and isolation valves, the radiation dose would not be substantially increased due to the loss of water. A repair team would be able to access the break location or piping isolation valves and either isolate the break or effect temporary repairs. FESW level could then be restored to normal.

There would be no immediate urgency to restore the level. No release of contamination is associated with this event. Active FESW cooling would be lost during this accident, but a test conducted during July 1993 with normal water level in the FESW indicates that considerable time is available to take action. This test, as documented in LACBWR Technical Report, LAC-TR-137, showed that with all cooling and coolant circulation to the pool isolated, FESW water temperature increased from 80°F to only 114°F in 15.5 days. This test was terminated at 114°F to limit increasing radioactivity in the pool water. Extrapolation of the data indicated the temperature would have stabilized at approximately 150°F. Due to the smaller water volume to act as the heat sink in the FESW pipe break accident the initial heat up rate of the FESW water would be approximately twice as great as that during the 1993 test. The heat removal rate from the FESW at a given temperature will be reduced somewhat since the wetted wall area is reduced by approximately 42%. The heat removal at a given temperature, by evaporation of the FESW coolant and condensation on the FESW cover and walls, will be essentially unchanged. Since heat removal rate increases rapidly as the temperature in the FESW increases, engineering judgment indicates the temperature in the FESW will stabilize at a temperature somewhat above 150°F, but boiling is not expected to occur. The total heat source in the FESW is only about 12.2 kW.

As with the loss of FESW cooling event, if water is added to the FESW, any consequences of water heat up can be delayed or prevented. Water can be added from the Demineralized Water System or the Overhead Storage Tank.

9.6 UNCONTROLLED WASTE WATER DISCHARGE

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

If the liquid in the tank is of high activity, the liquid waste monitor will alarm and the Auto Flow Control Valve (54-22-002) automatically will close, terminating the discharge. If the automatic valve does not close, an operator will try to close it from the Control Room. If it cannot be closed, an operator will close a local valve or secure the pump to terminate the discharge.

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

9.7 LOSS OF OFFSITE POWER

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

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

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

If neither EDG can be started, neither FESW or CCW pump can run. The consequences again are the same as a Loss of FESW Cooling Event. Some instrumentation will be lost immediately and the rest will be lost if packaged uninterruptible power supplies (UPS) are depleted. The operator would have to check the FESW locally periodically.

As discussed in Section 9.4, the fuel pool heatup test conducted in 1993 indicated that the temperature of the pool water would stabilize at less than boiling. Therefore, no immediate action needs to be taken and sufficient time is available to take corrective actions to restore power.

9.8 SEISMIC EVENT

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

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

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

9.9 WIND AND TORNADO

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

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

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

9.10 REFERENCES

1) NRC Letter, Ziemann to Linder, dated February 4, 1980.
2) NRC Letter, Reid to Madgett, dated October 22, 1975.
3) DPC Letter, Taylor to Document Control Desk, LAC-12377, dated September 29, 1987.\
4) DPC Letter, Linder to Paulson, LAC-10251, dated October 11, 1984.
5) NRC Letter, Zwolinski to Linder, dated January 16, 1985.
6) DPC Letter, Linder to Zwolinski, LAC-10639, dated March 15, 1985.
7) NRC Letter, Zwolinski to Taylor, dated September 9, 1986.
8) DPC Letter, Taylor to Zwolinski, LAC-12052, dated January 14, 1987.
9) NRC Letter, Bernero to Taylor, dated April 6, 1987.

LAC-TR-138

LACBWR

INITIAL

SITE CHARACTERIZATION SURVEY

FOR SAFSTOR

By:

Larry Nelson
Health and Safety Supervisor

October 1995

Revised: November 2006

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

PLANT SYSTEMS INTERNAL RADIONUCLIDE INVENTORY - JANUARY 1988 - (cont'd)

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

ATTACHMENT 1

SPENT FUEL RADIOACTIVITY INVENTORY

Decay-Corrected to October 2006

Radionuclide	*Half Life (Years)*	*Activity (Curies)*	*Radionuclide*	*Half Life (Years)*	*(Curies)*
Ce-144	7.801 E-1	0.15	Sr-90	2.770 E + 1	7.18E5
Cs-137	3.014 E+1	1.08E6	Pu-241	1.429 E+1	4.58E5
Ru-106	1.008 E+0	3.84	Fe-55	2.700 E+0	4.27E3
Cs-134	2.070 E+0	618	Ni-59	8.000 E+4	287
Kr-85	1.072 E+1	3.45E4	Tc-99	2.120 E+5	276
Co-60	5.270 E+0	5.43E3	Sb-125	2.760 E+0	2.46
Pm-147	2.620 E+0	290	Eu-155	4.960 E+0	12.2
Ni-63	1.000 E+2	3.11E4	U-234	2.440 E+5	63.7
Am-241	4.329 E+2	3.63E4	Am-243	7.380 E+3	61
Pu-238	8.774 E+1	1.09E4	Cd-113m	1.359 E+1	6.84
Pu-239	2.410 E+4	8.83E3	Nb-94	2.000 E+4	15.9
Pu-240	6.550 E+3	7.15E3	Cs-135	3.000 E+6	14.0
Eu-154	8.750 E+0	911	U-238	4.470 E+9	12.2
Cm-244	1.812 E+1	1.76E3	Pu-242	3.760 E+5	8.58
H-3	1.226 E+1	191	U-236	2.340 E+7	6.32
Eu-152	1.360 E+1	197	Sn-121m	7.600 E+1	3.74
Am-242m	1.505 E+2	449	Np-237	2.140 E+6	2.19
			U-235	7.040 E+8	1.89
			Sm-151	9.316 E+1	1.31
			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 = 2.38 E6 Curies

ATTACHMENT 2

CORE INTERNAL/RX COMPONENT RADIONUCLIDE INVENTORY – OCTOBER 2006

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)	1,866	506	1,187	8	3,567
Control Rods (29)	412	39	717	8	1,176
Core Vertical Posts (52)	107	5	55	2	169
Core Lateral Support Structure	769	172	676	4	1,621
Steam Separators (16)	2,822	631	2,481	15	5,949
Thermal Shield	122	27	108	0.5	257
Pressure Vessel	29	8	9	--	46
Core Support Structure	545	122	479	3	1,149
Horizontal Grid Bars (7)	15	3	13	--	31
Incore Monitor Guide Tubes	26	2	537	3	568
Total	6,713	1,515	6,262	43.5	14,534
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 2006

Plant System	Nuclide Activity, in µCi				System Total µCi Content
	Fe-55	Alpha	Co-60	Cs-137	
CB Ventilation	13	--	135	110	258
Offgas - upstream of filters	*SYSTEM*	*REMOVED*			
Offgas - downstream of filters	*SYSTEM*	*REMOVED*			
TB drains	136	40	1,435	3,245	4,856
CB drains	304	3	3,207	1,558	5,072
TB Waste Water	29	7	304	78	418
CB Waste Water	1680	79	17,724	1,493	20,976
Main Steam	2080	290	21,944		24,314
Turbine	7	2	78	130	217
Primary Purification	712	12	7,512		8,236
Emergency Core Spray	*SYSTEM*	*REMOVED*			
Overhead Storage Tank	104	34	1,097	506	1,741
Seal Inject	13	4	135	36	188

ATTACHMENT 3

PLANT SYSTEMS INTERNAL RADIONUCLIDE INVENTORY – OCTOBER 2006 - (cont'd)

Plant System	Nuclide Activity, in µCi			System Total µCi Content
	Fe-55	Alpha	Co-60	
Decay Heat	800	490	8,440	9,730
Boron Inject	*SYSTEM*	*REMOVED*		
Reactor Coolant PASS	*SYSTEM*	*REMOVED*		
Alternate Core Spray	160	94	1,688	1,942
Shutdown Condenser	*SYSTEM*	*REMOVED*		
Control Rod Drive Effluent	1,200	720	12,660	14,580
Forced Circulation	12,000	7,000	126,600	145,600
Reactor Vessel and Internals	20,000	12,000	211,000	243,000
Condensate after beds & Feedwater	*SYSTEM*	*REMOVED*		
Condensate to beds	*SYSTEM*	*REMOVED*		

ATTACHMENT 3

PLANT SYSTEMS INTERNAL RADIONUCLIDE INVENTORY – OCTOBER 2006 - (cont'd)

Plant System	Nuclide Activity, in µCi				System Total µCi Content
	Fe-55	Alpha	Co-60	Cs-137	
Fuel Element Storage Well System	6,800	390	71,740		78,930
Fuel Element Storage Well - all but floor	10	5	110	2,985	3,110
Fuel Element Storage Well floor	208,000	7,600	2,194,400	26,609	2,436,609
Resin lines	1,040	100	10,972		12,112
Main Condenser	88,000	8,500	928,400		1,024,900