WILLIAM L. BERG
President and CEO

DAIRYLAND POWER
COOPERATIVE

March 12, 2014

In reply, please refer to LAC-14298

DOCKET NO. 50-409 and 72-046

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: Dairyland Power Cooperative
La Crosse Boiling Water Reactor (LACBWR)
Possession-Only License DPR-45
LACBWR Decommissioning Plan and Post-Shutdown Decommissioning Activities Report (D-Plan/PSDAR) Revision – March 2014

Enclosed is the latest revision of the LACBWR D-Plan/PSDAR. The information in the present format provides stakeholders a better understanding of the current status of the decommissioning effort at LACBWR.

All pages of the document are included with change bars used in the right-hand margin to indicate revision. These changes have been reviewed by both the plant Operations Review Committee and the independent Safety Review Committee.

If you have any questions concerning these changes, please contact Don Egge, LACBWR Plant/ISFSI Manager, at (608) 689-4207.

Sincerely,

William L Berg

William L. Berg, President and CEO

WLB:JBM:jkl

Enclosures: 1) LACBWR Decommissioning Plan and Post-Shutdown Activities Report, Revised March 2014
2) Description of Changes

FSME20
NMSS26

A Touchstone Energy® Cooperative

cc w/Enclosures:

John Hickman
Project Manager
U.S. Nuclear Regulatory Commission

Cynthia Pederson
Regional Administrator, Region III
U.S. Nuclear Regulatory Commission

Paul Schmidt
Manager, Radiation Protection Section
State of Wisconsin

STATE OF WISCONSIN)
)
COUNTY OF LA CROSSE)

Personally came before me this 13th day of March, 2014, the above named, William L. Berg, to me known to be the person who executed the foregoing instrument and acknowledged the same.

Laurie A. Engen
Notary Public, La Crosse County Wisconsin

My commission expires 5-25-14

LAURIE A. ENGEN
Notary Public
State of Wisconsin

LACBWR DECOMMISSIONING PLAN

AND

POST-SHUTDOWN DECOMMISSIONING ACTIVITIES REPORT

REVISED MARCH 2014

DESCRIPTION OF CHANGES

Cover Page	Update Decommissioning Plan and Post-Shutdown Decommissioning Activities Report (D-Plan/PSDAR) revision date.
Page i – ii	Table of Contents is revised to reflect the March 2014 revision of the D-Plan/PSDAR.
Page 1-4	Section 1.2, Significant Post-Shutdown Licensing Actions: A new paragraph is added at the end of the section describing changes contained in License Amendment 72 that was approved by the NRC and issued July 31, 2013.
	Section 1.3, Metal Removal: In first paragraph, the phrase, *"prior to 2013,"* is added in the second sentence to clarify that the total weight of metallic waste cited as being shipped (over 2 million pounds) is prior to the current metal removal phase.
Page 1-5	Section 1.3.4, Primary Purification System: System status is revised to indicate that system demolition is in progress.
	Section 1.3.7, Fuel Element Storage Well System: Description is revised to note that the system provided the means for wet storage of spent fuel. System status is revised to indicate that the storage well has been drained; surface contamination has been fixed within the well; and that the system is prepared for demolition.
Page 1-6	Section 1.3.8, Component Cooling Water System: Description is revised to note that CCW equipment in the Turbine Building has been removed. System status is revised to indicate demolition of remaining CCW piping in Reactor Building is in progress.
	Sections 1.3.10, Well Water System and Section 1.3.11, Demineralized Water System: Both sections are revised to remove description of well water being supplied to laundry equipment and demineralized water being supplied to dry cleaner. Contaminated laundry operations are no longer being performed.
Page 1-7	Section 1.3.14, Low Pressure Service Water System: Description is revised to reflect that only one LPSW pump is available; that system no longer supplies CCW system or HPSW system; and that system may be used for liquid waste discharge dilution.
	Section 1.3.15, High Pressure Service Water System: Description is revised to reflect that system is used for fire suppression and is supplied by Genoa 3.
	Section 1.3.17, Condensate System and Feedwater Heaters: System status is revised to indicate that three feedwater heaters have been removed.
Page 1-8	Former Section 1.3.18, Full-Flow Condensate Demineralizer System, Section 1.3.19, Steam Turbine, and Section 1.3.20, 60-Megawatt Generator: Information about these systems has been deleted because all components of these systems have been removed. Subsequent sections have been renumbered.

Section 1.3.19, Heating, Ventilation, and Air Conditioning (HVAC) System: Description is revised to reflect that two units in Reactor Building are no longer used for cooling. Units are only used for heating and air handling.

Page 1-9 Section 1.3.20, Liquid Waste System: Description is revised to note that liquid waste discharge is made to the Circulating Water system and diluted by either LPSW or Circulating Water outflow. Description is also revised to note that Spent Resin Receiving Tank holds all remaining resin at the plant that will be transferred for disposal.

Page 1-9 Former Section 1.3.24, Fuel Transfer Bridge: Information about this system has been deleted because all components of the system have been removed. Subsequent sections have been renumbered.

Section 1.3.21, Electrical Distribution System: Description of two diesel generators available for standby power is deleted. These components have been removed from service and prepared for demolition.

Page 1-10 Section 1.4, Buildings and Structures: Clarification is made in first paragraph in that all buildings and structures within the radiological controlled area will be demolished and disposed of with the exception of the Cribhouse. This is a change in decommissioning methodology because some potentially contaminated structures were previously assumed to be decontaminated and left intact.

Page 1-14 Section 1.6, Groundwater: Former second paragraph described development of the hydrogeological conceptual site model (CSM) using a bulleted list of activities. This paragraph is deleted as unnecessary because groundwater sampling has commenced and CSM has been refined.

Page 1-15 Section 1.6, Groundwater: In the next to last paragraph of the section, new information is added providing description of results of initial groundwater sampling performed in 2013.

Page 1-16 Section 1.7, ISFSI Decommissioning: In last paragraph of section, final sentence containing ISFSI decommissioning cost estimate is deleted as being redundant here. Decommissioning cost information is provided in Section 3.1.

Page 2-2 Section 2.0, Schedule: In paragraph at top of page 2-2, information related to activities performed during development of the License Termination Plan including the determination of the disposition of concrete structures and site end use is deleted. DPC Board of Directors have authorized a change in decommissioning methodology in that all buildings and structures within the radiological controlled area will be demolished and disposed of with the exception of the Cribhouse. This is a change in decommissioning methodology because some potentially contaminated structures were previously assumed to be decontaminated and left intact.

Page 3-1 And Page 3-2 Section 3.1, Decommissioning Cost Financing: At bottom of page 3-1 and top of page 3-2, information is added that describes a cost study update performed in March 2013 for the LACBWR plant. This cost study update included the cost of demolition and disposal of all buildings and structures within the radiological controlled area with the exception of the Cribhouse. Some of these buildings and structures were previously assumed to be decontaminated and left intact. This change in decommissioning methodology results in an increase in the plant decommissioning cost estimate in the range of $20 million over the previous estimate.

On page 3-2, a paragraph is added that describes a revision to the ISFSI decommissioning cost estimate performed in March 2013. This revision used as-built dimensions of the vertical concrete casks which resulted in a reduction in the cost estimate for decommissioning the ISFSI.

On page 3-2, in the final paragraph of Section 3.1, information is added concerning requirements for submitting the ISFSI Decommissioning Funding Plan every three years. To be included with this triennial funding plan submittal is the ISFSI Decommissioning Cost Estimate that also shall be revised every three years.

DAIRYLAND POWER COOPERATIVE

LA CROSSE BOILING WATER REACTOR

DECOMMISSIONING PLAN

AND

POST-SHUTDOWN DECOMMISSIONING ACTIVITIES REPORT

REVISED

MARCH 2014

REVIEWED BY Jeff McRill DATE 3/6/2014
Technical Support Engineer, Licensing

REVIEWED BY E Martin DATE 3/6/2014
Manager, Quality Assurance

APPROVED BY D L E DATE 3/6/14
Plant/ISFSI Manager

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1.0 DESCRIPTION OF PLANNED DECOMMISSIONING ACTIVITIES

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

The November 2012 revision of the Decommissioning Plan established information in a Post-Shutdown Activities Report (PSDAR) format to provide stakeholders a better understanding of the current status of the decommissioning effort at LACBWR and the planned dismantlement activities.

There are 333 spent fuel assemblies stored in five NAC-MPC dry cask storage systems at the onsite Independent Spent Fuel Storage Installation (ISFSI). DPC currently expects the fuel to remain onsite until a federal repository, offsite interim storage facility, or licensed temporary monitored retrievable storage facility is established and ready to receive LACBWR fuel.

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

1.1 SELECTION OF SAFSTOR

Effective August 28, 1996, the NRC's final decommissioning rule amended the regulations on decommissioning procedures. The rule clarified ambiguities in previous regulation, reduced unnecessary requirements, provided additional flexibility, and codified procedures and terminology that had been used on a case-by-case basis. The 1996 rule extended the use of the process described in 10 CFR 50.59, "Changes, Tests, and Experiments," to allow licensees to make changes to facilities undergoing decommissioning if determined that prior NRC approval was not required.

The "Generic Environmental Impact Statement (GEIS) on Decommissioning of Nuclear Facilities," NUREG-0586, Supplement 1, evaluates the environmental impact of three methods for decommissioning. The Supplement updates information in the 1988 GEIS and discusses the three decommissioning methods; a short summary of each follows:

DECON is the alternative in which the equipment, structures, and portions of a facility and site containing radioactive contaminants are removed or decontaminated to a level that permits the property to be released for unrestricted use shortly after cessation of operations.

SAFSTOR is the alternative in which the nuclear facility is placed and maintained in such condition that the nuclear facility can be safely stored and subsequently decontaminated (deferred decontamination) to levels that permit release for unrestricted use.

ENTOMB is the alternative in which radioactive contaminants are encased in a structurally long-lived material, such as concrete. The entombed structure is appropriately maintained and continued surveillance is carried out until the radioactivity decays to a level permitting unrestricted release of the property. This alternative would be allowable for nuclear facilities contaminated with relatively short-lived radionuclides such that all contaminants would decay to levels permissible for unrestricted use within a period on the order of 100 years. For a power reactor, the choice was either DECON or SAFSTOR. Due to some of the long-lived isotopes in the reactor vessel and internals, ENTOMB alone was not an allowable alternative under the original proposed rule.

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

The overriding factor affecting the decommissioning decision for LACBWR was that a federal repository was not expected to be available for fuel storage in the foreseeable future. With the fuel in the Fuel Element Storage Well, the only possible decommissioning option was SAFSTOR. Limited decontamination and dismantling of unused systems could be performed during this period.

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

The shutdown of LACBWR occurred before the full funding for DECON was acquired. The SAFSTOR period has permitted the accumulation of the full DECON funding. The majority of studies showed that while the total cost of SAFSTOR with delayed DECON was greater than immediate DECON, the present value was less for the SAFSTOR with delayed DECON option.

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

A second disadvantage of delaying the final decommissioning was that the people who operated the plant would not be available for the DECON period. When immediate DECON is selected, some of the experienced plant staff would be available for decommissioning and dismantlement activities. When SAFSTOR is chosen, efforts must be made to maintain excellent records to compensate for the lack of staff continuity.

The remaining factor was safety. As of October 2009, 24 power reactors have been shut down in the United States, 11 of which have been fully dismantled and decommissioned. Experience has shown that the process can be performed safely.

The NRC issued its Waste Confidence Decision in August 1984 as codified in 10 CFR 51.23. Amended in December 2010, the NRC has found "reasonable assurance that, if necessary, spent fuel generated in any reactor can be stored safely and without significant environmental impacts for at least 60 years beyond the expiration of that reactor's operating license at that reactor's spent fuel storage basin, or at either onsite or offsite independent spent fuel storage installations." Therefore, DPC's plan to maintain the spent fuel at LACBWR, until a federal repository, interim storage facility, or licensed temporary monitored retrievable storage facility is ready to accept the fuel, is acceptable from the safety standpoint, as well as necessary from the practical standpoint.

After evaluating the factors involved in selecting a decommissioning alternative, DPC decided to choose an approximate 30-50 year SAFSTOR period, followed by DECON. After 25 years in SAFSTOR and all spent fuel in dry cask storage at the ISFSI, LACBWR is now beginning the final decommissioning and dismantlement phase.

1.2 SIGNIFICANT POST-SHUTDOWN 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 Provisional 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 provided evaluation and approval of the proposed Decommissioning Plan, post-operating Technical Specifications, and license renewal to accommodate the SAFSTOR period for a term to expire March 29, 2031.

The Decommissioning Order was modified September 15, 1994, by Confirmatory Order to allow DPC 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.

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 LACBWR Decommissioning Plan was considered the PSDAR. The PSDAR public meeting was held on May 13, 1998.

License Amendment No. 71 was issued January 25, 2011, making changes to the LACBWR license Appendix A, Technical Specifications in support of the Dry Cask Storage Project. The amendment revised the definition of FUEL HANDLING, reduced the minimum water coverage over stored spent fuel from 16 feet to 11 feet, 6½ inches, and made a small number of editorial changes to clarify heavy load controls and reflect inclusion of the cask pool as part of an "extended" Fuel Element Storage Well. The intent of these changes was to facilitate efficient dry cask storage system loading operations and reduce overall occupational dose to personnel during these operations.

License Amendment No. 72 was issued July 31, 2013, revising certain license conditions and removing Technical Specification definitions, operational requirements, and specific design requirements for wet spent fuel storage that are no longer applicable with all spent fuel in dry cask storage. The changes removed administrative control requirements that have been relocated to the LACBWR Quality Assurance Program Description or have been superseded by regulation or other guidance. Changes to the body of the license reflected revision of and exemptions granted for the Physical Security Plan described in License Condition 2.C.(3). The body of the license was also revised to describe ongoing changes to the Fire Protection Program described in License Condition 2.C.(4). The amendment reduced Technical Specifications to simply two design feature items: one a description of the licensed facility, and the other a declaration that a maximum of 333 spent fuel assemblies are stored in 5 dry casks within an Independent Spent Fuel Storage Installation (ISFSI). The declaration of fuel storage at the ISFSI includes the commitment that spent fuel assemblies shall not be placed in the spent fuel pool.

1.3 METAL REMOVAL

Significant dismantlement has already been accomplished. Over 2 million pounds of metallic waste has been removed and shipped prior to 2013, excluding the reactor pressure vessel (RPV) and spent fuel storage racks.

Included in the scope of work during the RPV project was removal of irradiated hardware and all other Class B and Class C material. Waste stored in the FESW was processed and with other B/C waste (i.e., resins, filters, and waste barrel contents) was collected in three liners and shipped for disposal in June 2007. The RPV with head installed, internals intact, and 29 control rods in place was filled with low density cellular concrete. Attachments to the RPV were removed and all other appurtenances were cut. The RPV was removed from the Reactor Building and was also shipped for disposal in June 2007. Following placement of all spent fuel assemblies and fuel debris in dry cask storage at the ISFSI in September 2012, the storage racks and installed components were removed from the spent fuel pool.

As described in Section 2, "Schedule," metal removal is the initial phase of final LACBWR decommissioning following dry cask storage. The remaining systems subject to this metal removal effort are described below.

1.3.1 Forced Circulation System

The Forced Circulation system circulated water through the reactor to cool the core and controlled reactor power from 60 to 100 percent. The system had two pumps with 16-inch and 20-inch piping connected by nozzles to the lower head of the reactor vessel.

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. 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.

1.3.2 Seal Injection System

The Seal Injection system provided cooling and sealing water for the seals on the two Forced Circulation pumps and the 29 control rod drive units.

System Status: This system is drained and not maintained operational.

1.3.3 Decay Heat Cooling System

The Decay Heat Cooling system was a single high pressure closed loop containing a pump, cooler, and interconnecting piping used to remove core decay heat following reactor shutdown.

System Status: This system is drained and not maintained operational.

1.3.4 Primary Purification System

The Primary Purification system was a high pressure, closed loop system consisting of a regenerative cooler, purification cooler, pump, two ion exchangers and filters.

System Status: Ion exchanger resins have been removed. System demolition is in progress.

1.3.5 Alternate Core Spray System

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

System Status: System components continue to serve requirements of the HPSW system.

1.3.6 Gaseous Waste Disposal System

This system routed main condenser gasses through various components for drying, filtering, recombining, monitoring and holdup for decay.

System Status: This system, except for the underground gas storage tanks, has been removed.

1.3.7 Fuel Element Storage Well System

The storage well is a stainless steel lined concrete structure 11 feet by 11 feet by approximately 42 feet deep and provided the means for wet storage of spent fuel. The cooling system was connected to the well and consisted of two pumps, one heat exchanger, one ion exchanger, piping, valves, and instrumentation.

System Status: All spent fuel and fuel debris, installed components, and storage racks have been removed from the storage well. The system has been drained and surface contamination has been fixed within the well. The system is prepared for demolition.

1.3.8 Component Cooling Water System

The Component Cooling Water (CCW) system provided controlled quality cooling water to the various heat exchangers and pumps in the Reactor Building during plant operation serving as a barrier between radioactive systems and the river. Pumps, heat exchangers, piping, and components located in the Turbine Building have been removed.

System Status: Demolition of remaining CCW piping in the Reactor Building is in progress.

1.3.9 Hydraulic Valve Accumulator System

The function of the Hydraulic Valve Accumulator system was to supply the necessary hydraulic force to operate the five piston-type valve actuators, which operated the five rotoport valves in the Forced Circulation and Main Steam systems.

System Status: This system has been drained. The air compressors, water pumps, and other equipment have been electrically disconnected and are not maintained operational.

1.3.10 Well Water System

Water for this system is supplied from two sealed submersible deep well pumps that take suction through stainless steel strainers, and discharge into integrated pressure tanks. The system supplies water to the plant and office for sanitary and drinking purposes. It is used as cooling water for the two Turbine Building air-conditioning units and the heating boiler blowdown flash tank and sample cooler. The well water system supplies seal water for the Circulating Water pumps.

System Status: This system is maintained in continuous operation.

1.3.11 Demineralized Water System

The Virgin Water Tank provides stored high quality water to the Demineralized Water transfer pumps which distribute demineralized water throughout the plant. Water is demineralized in batches at G-3 and transferred to LACBWR. The Condensate Storage Tank and the Virgin Water Tank are two sections of an integral aluminum tank located on the Turbine Building office roof. The lower section is the Condensate Storage Tank and has a capacity of 19,100 gallons. The upper section is the Virgin Water Tank and has a capacity of 29,780 gallons.

System Status: The Demineralized Water system remains available for service as a source of water for the heating boiler. The Condensate Storage Tank has been drained.

1.3.12 Overhead Storage Tank

The Overhead Storage Tank (OHST) is a 45,000 gallon tank located at the top of, and is an integral part of, the Reactor Building. The OHST served as a reservoir for water used to flood the Fuel Element Storage Well, cask pool, and upper cavity during cask loading operations. During operation, the OHST acted as a receiver for rejecting refueling water using the Primary Purification system. The OHST also 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 OHST has been drained.

1.3.13 Station and Control Air System

There are two single-stage positive displacement lubricated type compressors. The air receivers act as a volume storage unit for the station. The air receiver outlet lines join to form a header for supply to the station and the control air systems.

System Status: This system is operated as needed.

1.3.14 Low Pressure Service Water System

The system is supplied by one vertical pump located in the Cribhouse through a duplex strainer unit.

System Status: This system is maintained available for periodic dilution of liquid waste discharges.

1.3.15 High Pressure Service Water System

The HPSW system supplies fire suppression water. HPSW system is cross-connected with the fire suppression system of Genoa 3 (G-3). Pressure in the system is maintained by G-3. The HPSW system supplies interior fire stations, sprinkler systems, and outside fire hydrants.

System Status: This system is maintained operational for fire protection.

1.3.16 Circulating Water System

Circulating water is drawn into the Cribhouse intake flume from the river by two pumps located in separate open suction bays. Each pump discharges into 42-inch pipe that join a common 60-inch pipe leading to the main condenser in the Turbine Building. At the condenser, the 60-inch pipe branches into two 42-inch pipe to the top section of the water boxes. Water enters the top section of the condenser tube side and is discharged from the bottom section tube side. The 42-inch condenser circulating water outlet lines tie into a common 60-inch line which discharges to the seal well from G-3 located approximately 600 feet downstream from the Cribhouse.

System Status: This system is maintained available for periodic dilution of liquid waste discharges.

1.3.17 Condensate System and Feedwater Heaters

The Condensate system took condensed steam from the condenser hotwell and delivered it under pressure to the suction of the reactor feed pumps.

System Status: This system has been removed with the exception of the Condensate Storage Tank which has been drained.

1.3.18 Turbine Oil and Hydrogen Seal Oil System

The Turbine Oil system received cooled oil from the lube oil coolers to supply the necessary lubricating and cooling oil to the turbine and generator bearings, exciter bearings, and exciter reduction gear.

System Status: This system has been removed with the exception of the drained clean and dirty oil tanks.

1.3.19 Heating, Ventilation, and Air-Conditioning (HVAC) Systems

The Reactor Building ventilation system utilizes two 12,000-cfm air handling units for drawing fresh air into the building and for circulating the air throughout the building. Each air handling unit inlet is provided with a filter box assembly, face and bypass dampers; and one 337,500-Btu/hr capacity steam coil that is used when heating is required. Air enters the building through openings between and around the bi-parting door sections and is exhausted from the building by action of the stack blowers. Additional exhaust flow is available using a centrifugal exhaust fan that has a capacity of 6,000 cfm at 4 inches of water static pressure. The exhaust fan and building exhaust air discharge through two series 20-inch dampers to the Reactor Building ventilation outlet plenum connected to the tunnel. A 20-inch damper is also provided for recirculation of the exhaust fan discharge. The exhaust system is provided with conventional and high-efficiency filters and with a particulate radiation monitor system.

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

The exhaust air from the Reactor Building and the Waste Treatment Building is discharged into the tunnel connecting the Waste Treatment Building, the Reactor Building, and the Turbine Building to a plenum at the base of the stack. The stack is 350 feet high and is of structural concrete with an aluminum nozzle at the top.

The Turbine Building heating system provides heat to the turbine and machine shop areas through unit heaters and through automatic steam heating units. The Control Room Heating and Air-Conditioning unit serves the Control Room and Electrical Equipment Room. The office area and laboratory are provided with a separate multi-zone heating and air-conditioning unit.

The heating boiler is a Cleaver-Brooks, Type 100 Model CB-189, 150-hp unit. At 150 psig, the boiler will deliver 6,375,000 Btu/hr. The boiler fuel is No. 2 fuel oil. The oil is supplied by and atomized in a Type CB-1 burner which will deliver 45 gph.

System Status: HVAC systems are maintained operational.

1.3.20 Liquid Waste Collection Systems

The Turbine Building Liquid Waste system collects the liquid waste from the Turbine Building, the Waste Treatment Building, the gas storage tank vault, and the tunnel area in two storage tanks (4500 gallons and 3000 gallons) located in the tunnel between the Reactor Building and the Turbine Building.

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

After a tank's contents are recirculated, a sample is drawn from the tank and analyzed for radioactivity concentrations prior to discharge. The liquid waste is then discharged to the Circulating Water system and diluted by LPSW or Circulating Water outflow.

The spent resin receiving tank holds the remaining spent resin that was used at the plant. The resin will be transferred to an approved shipping container for disposal.

System Status: Liquid Waste Collection systems are maintained operational.

1.3.21 Electrical Distribution System

69-kV power is supplied to the reserve auxiliary transformer (RAT) located in the LACBWR switchyard through a three-phase air-disconnect switch and three 30-amp, 69-kV fuses. Reserve Feed Breakers supply the 2400-V Bus 1A and Bus 1B from the 69/2.4-kV RAT. The 2400/480-V Auxiliary Transformers 1A and 1B receive power from the 2400-V Buses 1A and 1B through breaker 252AT1A from Bus 1A to Transformer 1A, and breaker 252AT1B from Bus 1B to Transformer 1B. The auxiliary transformers supply the 480-V Buses 1A and 1B through Main Feed Breakers 452M1A for Bus 1A and 452M1B for Bus 1B.

The 480-V buses supply larger equipment directly and motor control centers (MCCs) that furnish power to motors and equipment connected to them including 120-V AC Distribution Panels. Regular lighting cabinets are supplied from 480-V Buses 1A and 1B.

The 125-V DC distribution system supplies DC power for equipment and AC breaker operations. The Diesel Building Battery Charger, Generator Plant Battery, and Generator Plant Battery Charger remain as sources of DC power. The Diesel Building Battery Charger provides the normal DC supply with the Generator Plant Battery as the reserve supply. The battery floats on the system maintaining a full charge, and provides DC power in the event of a loss of AC power to the battery charger or failure of the charger. Due to age, the Generator Plant Battery Charger is maintained available as a standby unit.

System Status: The Electrical Power Distribution system is maintained operational.

1.3.22 Post-Accident Sampling Systems

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

System Status: The Stack PASS is maintained in continuous operation ensuring flow for AMS-4 monitoring of stack effluent particulate. The Reactor Coolant PASS has been removed. The Reactor Building Atmosphere PASS remains in place. With all spent fuel transferred to dry cask storage at the ISFSI, Post-Accident Sampling Systems are no longer required.

1.4 BUILDINGS AND STRUCTURES

Located within the radiological controlled area of LACBWR are the following buildings and structures. These buildings and structures, with the exception of the Cribhouse, will be demolished and disposed of.

- Reactor Building
- Turbine Building and Turbine Office Building
- Waste Treatment Building
- Low Specific Activity (LSA) Storage Building
- Cribhouse
- Maintenance Eat Shack
- Underground Gas Storage Tank Vault
- 1B Diesel Generator Building
- Ventilation Stack

1.4.1 Reactor Building

The Reactor Building is a right circular cylinder with a hemispherical dome and semi-ellipsoidal bottom. It has an overall internal height of 144 feet and an inside diameter of 60 feet, and it extends 26'-6" below grade level. The shell thickness is 1.16 inch, except for the upper hemispherical dome which is 0.60 inch thick.

The building contained most of the equipment associated with the nuclear steam supply system, including the reactor vessel and biological shielding. The interior of the shell is lined with a 9-inch-thick layer of concrete to an elevation of 727'-10" 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 feet thick. This support runs from the bottom of the semi-ellipsoidal head at about elevation 612'-4" to an elevation of 621'-6". The 232 piles that support the containment structure are driven deep enough to support over 50 tons per pile.

The containment bottom head above elevation 621'-6" and the shell cylinder from the bottom head to approximately 9 inches above grade elevation 639 feet are enveloped by reinforced concrete laid over a ½ inch thickness of pre-molded expansion joint filler. The reinforced concrete consists of a lower ring, mating with the pile capping concrete. The ring is approximately 4½ feet thick at its bottom and 2½ feet thick at a point 1½ feet below the top due to inner surface concavity. The ring then tapers externally to a thickness of 9 inches at the top (elevation 627'-6") and extends up the wall of the shell cylinder to elevation 639'-9".

The shell includes two airlocks. The personnel airlock connects the Reactor Building to the Turbine Building. The airlock is 21'-6" long between its two rectangular doors that measure 5'-6" by 7'. The Reactor Building is also equipped with an emergency airlock, which is 7 feet long and 5 feet in diameter, with two circular doors of 32½-inch diameter (with a 30-inch opening). Both airlocks are at elevation 642'-9" and lead to platform structures from which descent to grade level can be made.

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

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 feet below grade level either at the northwest quadrant or at the northeast quadrant and enter the pipe tunnel connecting the Turbine Building, Reactor Building, stack, Waste Treatment Building, and the underground gas storage tank vault.

A 45,000-gallon 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 provided a source of water inventory for fuel handling operations and the FESW.

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 was stored underwater in racks in the bottom of the FESW located adjacent to the reactor biological shielding in the Reactor Building. The storage rack system was a two-tier configuration such that each storage location was capable of storing two fuel assemblies, one above the other.

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'-8". The width of the upper 24'-8" of the opening is 16'-9¼" and the width of the lower 34' of the opening is 10'-6". The opening is closed by a weather tight, insulated, roll-up, bi-parting door.

For dry cask storage operations, the Reactor Building mezzanine floor north was reinforced by adding steel struts beneath a cantilevered section of the floor. In order to provide sufficient water coverage over the spent fuel assemblies during movement into the TSC from the FESW, a water-tight removable gate, 16'-9" high by 9'-4" wide, was installed in the bio-shield opening above approximate elevation 679'-3" extending to elevation 696 feet. The cask pool gate was supported by a 12' high structure installed at elevation 667 feet. The cask pool gate was designed with inflatable pneumatic seals having a defined acceptable leakage rate. Appropriate interfacing modifications to the bio-shield liner at the edges of the opening were installed to ensure water retention in the area between the upper cavity liner and the cask pool gate. The cask pool gate storage stand supported the 6-ton cask pool gate when not in use.

The 10' high by 10' inner diameter cask pool was installed at elevation 669'-3" atop a 20'-10" high support structure attached to the reactor support cylinder at elevation 648'-5". The cask pool had a 16½" wide horizontal flange welded to the top of the shell, the outer circumference of which was tied into the existing upper cavity liner using L-shaped stainless steel angle at approximate elevation 679'-3". This arrangement provided a barrier to prevent water in the upper cavity area above the cask pool from leaking around the outside of the cask pool into the cavity below.

1.4.2 Turbine Building and Turbine Office Building

The Turbine Building contained the steam turbine and generator, main condenser, electrical switchgear, and other pneumatic, mechanical and hydraulic systems and equipment required for a complete power plant. A 30/5-ton capacity, remote-operated overhead electrical traveling crane spans the Turbine Building. The crane has access to major equipment items located below the floor through numerous hatches in the main floor. The Turbine Building is 104.5 feet by 79 feet and 60 feet tall.

The Turbine Office Building contained offices, the Control Room, locker room facilities, laboratory, shops, counting room, personnel change room, 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 Turbine Office Building is 110 feet by 50 feet and 45 feet tall.

1.4.3 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 WTB 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 temporary storage space for processed dry active waste 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 WTB is 34 feet by 42 feet and 20 feet tall. The WTB basement floor is at elevation 630 feet and has a 3-feet deep sump with 8-inch thick walls and bottom which extend to a depth of 626'-4".

The Low Specific Activity (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. The building has the capacity for 500 DOT17H-55 gallon drums of waste. No liquids are stored in this building and there are no effluent releases from this building during normal use. The LSA Building is 27 feet by 80 feet and 15 feet tall.

1.4.4 Cribhouse

The Cribhouse is located on the bank of the Mississippi River to the west of the plant and through its intake structure, provided river water used in the various plant systems. The Cribhouse contains the diesel-driven High Pressure Service Water pumps, Low Pressure Service Water pump, and the Circulating Water pumps. The Cribhouse is 35 feet by 45 feet and 15 feet tall.

1.4.5 Maintenance Eat Shack

The Eat Shack is a 20 feet by 40 feet and 15 feet tall steel-sided building with windows constructed over a concrete slab.

1.4.6 Underground Gas Storage Tank Vault

The gas storage tank vault is a 29'-6" by 31'-6" (outside dimensions) underground concrete structure with 14-feet high walls and 2-feet thick floors, walls, and ceiling. The vault is 3-feet below grade elevation of 639 feet and with sump extends to a depth of 22 feet or elevation 617 feet. Two 1,600 cubic feet tanks are located in the underground gas storage tank vault. The tanks had the capability to store radioactive gases until such time that they were batch released via the stack. The tanks remain in place below grade along with the associated piping.

1.4.7 1B Diesel Generator Building

The 1B Diesel Generator Building is attached to the southeast corner of the Turbine Building and contains the Electrical Equipment Room, Diesel Generator Room, and an empty Battery Room. The building is constructed of concrete block and steel beams and braces. The building is L-shaped having largest dimensions of 30'-10" by 37'-11" and 13 feet tall.

1.4.8 Ventilation Stack

The LACBWR ventilation stack is a 350-feet high, tapered, reinforced concrete structure with an outside diameter of 7.19 feet at the top and 24.719 feet at the base. The wall thickness varies from 15 inches at the bottom to 6 inches at the top. The 4-feet thick foundation mat rests on a pile cluster of 78 piles. The foundation mat is 39'-8" square formed without triangular sections of equal 11'-7½" sides on the southeast and southwest corners.

1.5 DECOMMISSIONING WASTE DISPOSAL

1.5.1 System Classification

As a lesson learned from the dismantlement of other nuclear facilities, removal costs and removal contingency percentages vary, depending upon whether the system is a contaminated system or can be classified as a potentially clean system. To more accurately determine LACBWR's final cost and to better apply reprocessing, labor, and burial costs, LACBWR dismantlement activities are broken into categories based on contamination levels as follows:

- > 20 mRem/hr general dose rate
- < 20 mRem/hr general dose rate
- Potentially clean, < 0.1 mRem/hr general dose rate

1.5.2 Shipping

Transportation of material from LACBWR will be made by either truck or rail. In most cases, truck transport will be utilized because it is readily available and normally less expensive. Rail will be used to transport contaminated concrete and soils because of the large volumes of material needing transport.

1.5.3 Radioactive Waste Recycling

The use of radwaste processors has been found to consistently decrease the cost for direct burial of decommissioning waste. LACBWR has been a long-time user of radwaste processors. Past experience of radioactive waste shipments from LACBWR, sent for processing, has shown that of the total material processed approximately 50% has been sent to burial. LACBWR will continue to process all of the Class A material removed, except concrete and soil, to reduce the total volume of waste material needing burial.

A unique aspect in the use of a radwaste processor is that, by signing an agreement for the processing, set rates are established between LACBWR and the processor. The processor's contracted rates cover both the processing and burial of materials sent to their facilities. These radwaste processing charges are fixed by contract.

1.5.4 Burial

All LACBWR Class A metallic waste and dry active waste (not concrete or soil) will be sent to a processor and are not considered in the burial costs. Concrete and soil produced in the decommissioning of LACBWR will be shipped for direct burial. These shipments, due to volume, will be made by rail when possible. Class B and C waste, will not be created as a burial site is not available.

1.6 GROUNDWATER

Groundwater characterization is a requirement of decommissioning nuclear power plants and although each station may have varying degrees of investigations, techniques, and modeling efforts, the process is similar. Haley & Aldrich, Inc. was contracted to build the conceptual site model (CSM) for the LACBWR site with respect to the potential release of radiological and chemical materials to the environment. The Hydrogeological CSM was the first step to better understand both groundwater flow regimes as well as groundwater quality, with respect to radionuclides associated with LACBWR.

Data gathered through investigation were used to obtain a better understanding of the site's history and hydrogeological setting and were used to design a sampling program. Since these data will also be used to support license termination, they will provide hydrogeological information for a model such as RESRAD and to develop site Derived Concentration Guideline Levels (DCGLs).

In November 2012, five pairs of groundwater monitoring wells (10 wells total) were installed within the LACBWR radiological controlled area in the most-likely areas of potential release to determine if groundwater quality has been impacted. The paired wells were installed downgradient of the most likely areas where potential releases occurred and have sufficient spatial distribution so that groundwater flow rates and direction may be estimated. The paired wells were installed so that the shallow well intersects the water table and the deeper well is installed at depths approximately 20 to 30 feet below the shallow well. Soil samples were collected during the well installation to provide additional data points that will be able to support RESRAD, should DCGLs be needed during future decommissioning actions (i.e., if radiological contamination is present in soils and or groundwater).

Two rounds of groundwater samples were collected as part of the hydrogeological investigation. Samples were collected during the seasonal high water in June 2013 and then again during a seasonal low groundwater level in November 2013. Groundwater samples were collected using low flow methods at monitoring wells equipped with dedicated tubing.

Results of the initial groundwater sampling performed in 2013 indicate:

- The shallow aquifer has slower velocities and groundwater movement below the Turbine Building and faster groundwater movement outside and around the Turbine Building, suggesting some interference of the subsurface pilings associated with the building. Groundwater velocity data for the deep aquifer indicate less variability and lack the influence of subsurface disturbances.

- The most likely areas of interest (AOIs) where radionuclides could have been released to soils and groundwater include the Turbine Building waste collection system and the Underground Gas Storage Tank Vault and piping. No radionuclides were detected above background from the groundwater monitoring wells suggesting that these AOIs did not impact downgradient conditions. However, soils and groundwater directly below these areas have not yet been characterized.

- Groundwater analytical results did not report radionuclides at activities above background in any of the samples; historic site operations did not significantly impact groundwater quality downgradient of the potential AOIs.

The focus of the investigation is to characterize groundwater quality in support of NRC decommissioning requirements. However, monitoring wells are located and constructed using methods, such that if needed, they meet the data quality objectives of other programs that may come into play during the regulatory closure of the site.

1.7 ISFSI DECOMMISSIONING

The decommissioning plan for the ISFSI is based on information contained in the NAC-MPC FSAR, Section 2.A.4, "Decommissioning Considerations." The ISFSI will be decommissioned after the stored spent fuel is removed and transferred to the Department of Energy. NAC-MPC dry cask storage systems in use at the ISFSI are designated as MPC-LACBWR.

The principal elements of the MPC-LACBWR storage system are the vertical concrete cask (VCC) and the transportable storage canister (TSC). The VCC provides biological shielding and physical protection for the contents of the TSC during long-term storage. The VCC is not expected to become surface contaminated during use, except through incidental contact with other contaminated surfaces. Incidental contact could occur at the interior liner surface of the VCC, the top surface that supports the transfer cask during loading and unloading operations, and the pedestal of the VCC that supports the TSC. All of these surfaces are carbon steel, and could be decontaminated as necessary for decommissioning. A ¼-inch stainless steel plate is placed on the carbon steel pedestal of the MPC-LACBWR VCC to separate it from the stainless steel TSC bottom. Contamination of these surfaces is expected to be minimal, since the TSC is isolated from spent fuel pool water during loading in the pool and the transfer cask is decontaminated prior to transfer of the TSC to the VCC. Activation of the VCC carbon steel liner, concrete, support plates, and reinforcing bar could occur due to neutron flux from the stored fuel. Since the neutron flux rate is low, only minimal activation of carbon steel in the VCC is expected to occur.

Decommissioning of the VCC would involve the removal of the TSC and the subsequent disassembly of the VCC. It is expected that the concrete would be broken up, and steel components segmented to reduce volume. Any contaminated or activated items are expected to qualify for near-surface disposal as low specific activity material.

The TSC is designed and fabricated to be suitable for use as a waste package for permanent disposal in a deep Mined Geological Disposal System, in that it meets the requirements of the DOE MPC Design Procurement Specification. The TSC is fabricated from materials having high long-term corrosion resistance, and the TSC contains no paints or coatings that could adversely affect the permanent disposal of the TSC. As a result, decommissioning of the TSC would occur only if the spent fuel contained in the TSC had to be removed. Decommissioning would require that the closure welds at the TSC closure lid and port covers be cut, so that the spent fuel could be removed. Removal of the contents of the TSC would require that the TSC be returned to a spent fuel pool or dry unloading facility, such as a hot cell. Closure welds can be cut either manually or with automated equipment, with the procedure being essentially the reverse of that used to initially close the TSC.

The LACBWR ISFSI storage pad, fence, and supporting utility fixtures are not expected to require decontamination as a result of use of the MPC-LACBWR system. The design of the VCC and TSC precludes the release of contamination from the contents over the period of use of the system. Consequently, these items may be reused or disposed of as locally generated clean waste.

The decommissioning plan for the ISFSI is to dispose of the five VCCs and the 32' x 48' x 3' concrete storage pad.

2.0 SCHEDULE

The current schedule for decommissioning activities at LACBWR is depicted in Figure 2.1. 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 SAFSTOR. License renewal granted at the same time accommodated the proposed SAFSTOR period for a term to expire March 29, 2031. At the time of the original Decommissioning Plan in 1987, DPC anticipated the plant would be in SAFSTOR for a 30-50 year period.

To make better use of resources during the SAFSTOR period, some incremental decommissioning 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 2008, approximately 2 million pounds of material related to the removal of unused components or whole systems, completed in over 100 specific approved changes to the facility, has been shipped for processing and disposal. This total does not include reactor vessel and B/C waste disposal.

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.

In 2007, DPC began efforts to place an ISFSI on site by commencing the Dry Cask Storage Project. An on-site ISFSI was the available option that provided flexibility for license termination of the LACBWR facility. With respect to the federal repository option, a marker for transport of spent fuel offsite has also been added to Figure 2.1 as best available information can provide.

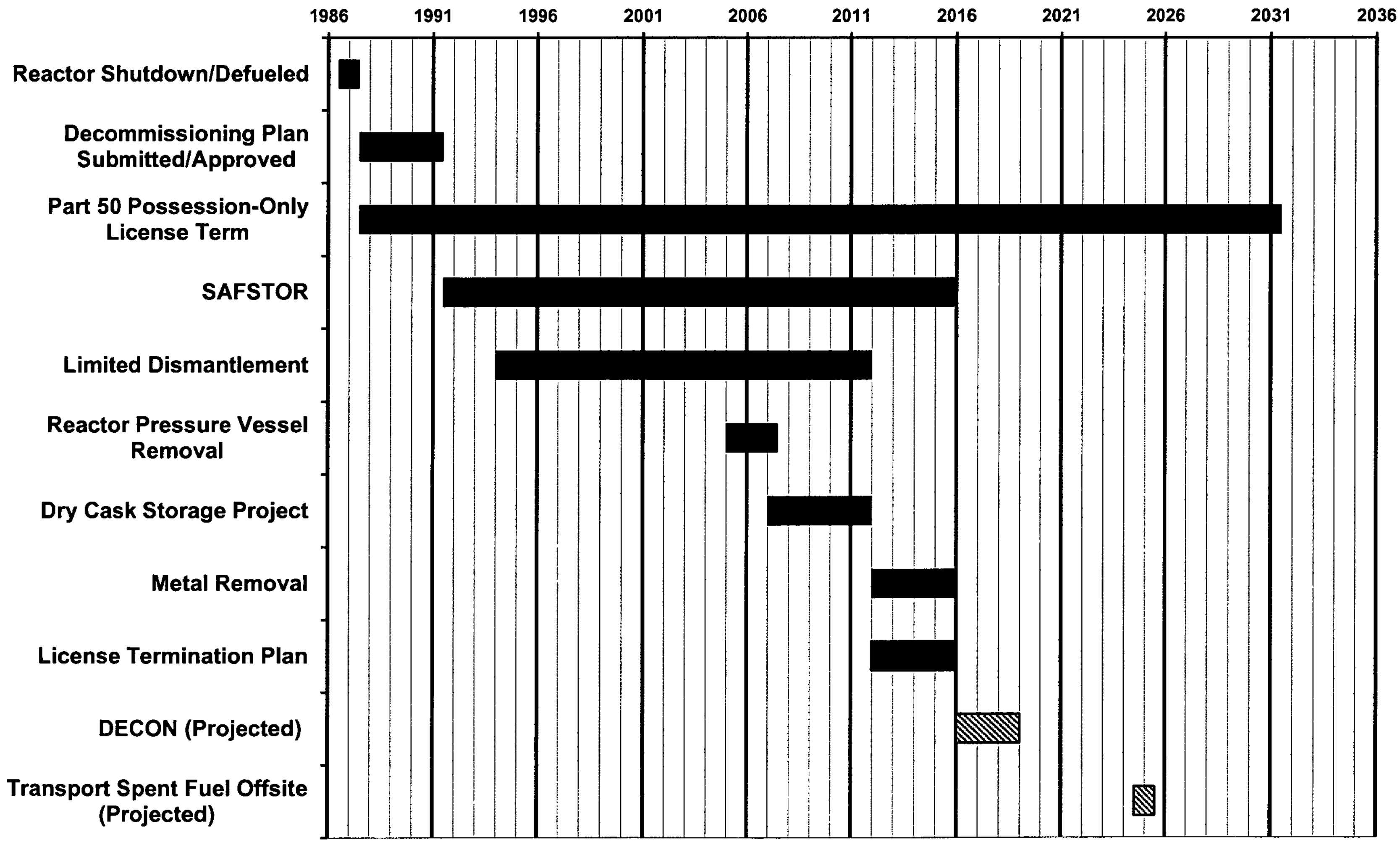
DPC Staff completed an extensive review and analysis of the comparative costs and benefits of the current decommissioning schedule and various accelerated schedules. From this analysis, the DPC Board of Directors approved accelerating the removal of radioactive metal from the LACBWR facility. By letter dated December 7, 2010, DPC gave notification to the NRC of a change in schedule that would accelerate the decommissioning of the LACBWR facility starting with a 4-year period of systems removal beginning in 2012. This activity will include the removal for shipment of large bore (16 and 20-inch) reactor coolant piping and pumps of the Forced Circulation system and other equipment once connected to the reactor pressure vessel or primary system such as Control Rod Drive Mechanisms, Decay Heat, Primary Purification, Seal Injection, and Main Steam.

This metal removal phase of decommissioning activity does not result in significant environmental impacts and has been reviewed as documented in the "Generic Environmental Impact Statement (GEIS) on Decommissioning of Nuclear Facilities," NUREG-0586, Supplement 1, November 2002. The GEIS characterizes the environmental impacts resulting from metal removal as generic and small.

DPC's review and analysis found that the Nuclear Decommissioning Trust (NDT) was sufficiently funded to allow dismantlement to begin in 2012 immediately after spent fuel removal was completed. Costs of the metal removal project will be funded from the NDT. DPC's approved strategy requires continuing evaluation of the costs of the decommissioning activity as it progresses. During this time the LTP will be formulated and will include an updated site-specific estimate of remaining decommissioning costs. DPC's decommissioning strategy for LACBWR with accelerated systems removal provides flexibility in that provisions are afforded to evaluate the costs and benefits of alternative methodologies for concrete removal, and delay LTP implementation if necessary to assure adequate NDT funds are available for the final decommissioning process. Figure 2.1 depicts the revised schedule.

The Dry Cask Storage Project established an ISFSI on the LACBWR site under the general license provisions of 10 CFR 72, Subpart K. The ISFSI is located 2,232 feet south-southwest of the Reactor Building center. The ISFSI is used for interim storage of LACBWR spent fuel in the NAC International, Inc. (NAC) Multi-Purpose Canister (MPC) System. The ISFSI contains all LACBWR spent fuel in five NAC-MPC dry cask storage systems. Cask loading and transport operations were completed on September 19, 2012, when the fifth and final dry storage cask was placed on the ISFSI pad.

FIGURE 2.1
LACBWR SCHEDULE



3.0 ESTIMATE OF EXPECTED DECOMMISSIONING COSTS

3.1 DECOMMISSIONING COST FINANCING

In late 1983, the DPC Board of Directors resolved to provide resources for the final dismantlement of LACBWR. DPC began making deposits to a decommissioning fund in 1984. The Nuclear Decommissioning Trust (NDT) was established in July 1990 as an external fund outside DPC's administrative control holding fixed income and equity investments.

The cost of DECON was based on the selection of unrestricted use as the criteria 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, NDT funds were approximately $83.4 million. NDT funds for B/C waste and RPV removals, 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.

A cost study update was completed in November 2010 to more accurately assess future costs of the remaining dismantlement needed and to facilitate DPC decommissioning and license termination planning. This update placed the cost to complete decommissioning at $67.8 million in Year 2010 dollars. During this process, ISFSI decommissioning costs were identified uniquely as a specific item and estimated to be $1.6 million in Year 2010 dollars. The DPC Board of Directors has established an external funding mechanism for ISFSI decommissioning costs in accordance with 10 CFR 72.30 to assure adequate funds will be available for the final decommissioning cost of the LACBWR ISFSI.

A cost study update was completed in March 2013 for the LACBWR plant. During the revision to the cost study, some potentially contaminated structures previously assumed to be decontaminated and left intact were evaluated for demolition and disposal. This change in decommissioning methodology to demolition and disposal of structures, in lieu of decontamination of structures, resulted in an increase in the LACBWR plant decommissioning cost estimate in the range of $20 million over the previous November 2010 decommissioning cost estimate of $67.8 million. The March 2013 cost study update included the cost of demolition and disposal of the LACBWR stack, Turbine and Turbine Office Buildings, Waste

Treatment Building, and Underground Gas Storage Tank Vault structure. This update placed the cost to complete plant decommissioning at $90.7 million in Year 2013 dollars. The DPC Board of Directors formally adopted the change in decommissioning methodology to demolition and disposal of potentially contaminated structures and authorized adjustments to decommissioning funding be made as necessary.

The ISFSI Decommissioning Cost Estimate was revised in March 2013 to reflect the MPC-LACBWR as-built vertical concrete cask (VCC) dimensions. These VCC dimensions differ from those used to establish the ISFSI Decommissioning Cost Estimate in 2010. Use of the as-built VCC dimensions resulted in a reduction in the volume of concrete to be disposed of. The cost for ISFSI decommissioning was estimated to be $1,435,626 in Year 2013 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 ISFSI 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. At least every five years, the plant decommissioning cost estimate will be revised in order to assure adequate funds will be available at the time of final decommissioning.

In accordance with 10 CFR 72.30(c), the ISFSI Decommissioning Funding Plan shall be reviewed, revised, and submitted to the NRC every three years. The triennial ISFSI Decommissioning Funding Plan submittal shall include the ISFSI Decommissioning Cost Estimate that also shall be revised every three years.

3.2 DRY CASK STORAGE COST FUNDING

10 CFR 50.54(bb) requires the establishment of a program by which the licensee intends to manage and provide funding for spent fuel storage at the reactor following permanent cessation of operation until which time the spent fuel is transferred to the Secretary of Energy for its ultimate disposal in a repository.

Pursuant to 10 CFR 50.54(bb), DPC promulgated the following spent fuel management funding plan that is now applicable to dry cask storage of the LACBWR spent fuel.

Independent of costs for dry cask storage of the LACBWR spent fuel; DPC has established the NDT and reports annually to the NRC the status of NDT funds. DPC understands that none of the funds in the NDT may be used for dry cask storage and operation of the ISFSI.

DPC will fund the expense of dry cask storage costs from the annual operating and maintenance budget. As part of generation expenses, dry cask storage 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 dry cask storage expenses as incurred.

DPC has found no need to separately fund dry cask storage costs outside the regular operating and maintenance budget. Dry cask storage costs are relatively small compared to DPC's annual operating and maintenance costs for generation and transmission facilities, and DPC will continue the long-standing policy of recovering dry cask storage costs as part of regular rates.

4.0 PLANT POST-FUEL ACCIDENT ANALYSIS

4.1 OVERVIEW

This section presents the results of an analysis (Reference 4.8.1) of postulated accidents that reflect the significantly reduced non-ISFSI radiological source term as compared to the LACBWR source term during plant operations. With consideration for the current stage of LACBWR decommissioning and with spent nuclear fuel now stored in the ISFSI, this analysis confirms that the minimal radioactive material resulting from LACBWR operation and remaining on the LACBWR site is insufficient for any potential event to result in exceeding dose limits or otherwise involving a significant adverse effect on public health and safety.

The analysis considers the spontaneous release of the (non-ISFSI-related) radioactive source term remaining at the LACBWR site in a form and quantity immediately releasable through the:

- Airborne pathway; and
- Liquid discharge pathway.

The airborne release and one of the liquid release events considered in the analysis are non-mechanistic in that there are no credible phenomena that could reasonably be postulated to cause such releases. However, these events are analyzed and conservative assumptions for other credible liquid release events are selected to bound any remaining decommissioning events that can still be postulated considering the current stage of LACBWR decommissioning.

4.2 RADIONUCLIDE RELEASE LIMITS APPLIED IN ANALYSIS

4.2.1 Limits Applied to Postulated Airborne Release

The following regulatory limits were considered in the analysis of a postulated airborne release:

1. The limits of 10 CFR 100.11 that specify that the total radiation dose to an individual at the exclusion area boundary for two hours immediately following onset of a postulated fission product release shall not exceed 25 rem (whole body) and 300 rem (thyroid; see Section 4.3.2.6).

2. The EPA protective action guidelines (PAGs – Reference 4.8.2) that specify the potential offsite dose levels at which actions should be taken to protect the health and safety of the public. The EPA PAG limits include a total effective dose equivalent (TEDE) of 1 rem.

The EPA PAGs are limiting values for the LACBWR post-fuel accident analysis. This conclusion is based on the sum of the effective dose equivalent resulting from exposure to external sources and from the committed effective dose equivalent incurred from the significant inhalation pathways during the early phase of an event. As detailed further in Section 4.4, this analysis demonstrates that there is insufficient releasable radioactive contamination remaining on the LACBWR site for reasonably conceivable radiological accident scenarios that could result in exceeding the EPA PAGs.

4.2.2 Limits Applied to Postulated Liquid Releases

The LACBWR analysis conservatively applies the normal effluent concentration limits of 10 CFR 20, Appendix B, Table 2, Column 2, to the event scenarios involving release of bulk radioactive liquids. As detailed further in Section 4.4, this analysis demonstrates that there is no reasonable likelihood that a postulated radioactive liquid release event could result in exceeding the normal effluent concentration limits of 10 CFR 20, Appendix B (Reference 4.8.3).

4.3 POST-FUEL ACCIDENT ANALYSIS ASSUMPTIONS

4.3.1 Assumptions – Remaining Non-ISFSI-Related Radioactive Source Term

With the spent nuclear fuel stored in the LACBWR ISFSI, the amount of (non-ISFSI-related) radioactive contamination conservatively assumed in the analysis to remain at the LACBWR site bounds the decreasing amounts present as decommissioning progresses and is completed. Potential sources of non-ISFSI radioactivity that remain at LACBWR include the following:

1. Radioactivity on surfaces of plant structures, systems, and components (SSCs);
2. Sealed and unsealed sources used for instrument calibration;
3. Filters used for liquid radwaste cleanup;
4. Assorted tools and equipment used to perform decommissioning activities; and
5. Radioactive waste containers stored awaiting shipment.

For purposes of the LACBWR post-fuel accident analysis, the radioactivity on plant surfaces is assumed to reasonably represent the non-ISFSI radioactive source term remaining at the LACBWR site (i.e., the other identified potential sources are negligible or are already accounted for as part of plant surface contamination). Specifically, sealed sources are designed to prevent the release of the contents and are not considered in this analysis to be a potential source of releasable radioactive material. Unsealed sources remaining at LACBWR are of extremely low radioactivity levels, such that they do not contribute significantly to the total releasable source term considered in the analysis. Filters are used to remove radioactive material from radioactive liquids generated from decommissioning activities. The radioactive material in these filters is material that is already accounted for above when considering the contamination contained on plant surfaces. Thus, liquid radioactive waste filters do not result in additional releasable source term beyond that already considered.

Radioactive material on or within tools and equipment used at LACBWR is of extremely low radioactivity levels, such that this material constitutes only a small fraction of the radioactivity on plant surfaces. Thus, tools and equipment do not contribute significantly to the total releasable source term considered in the analysis. Finally, radioactive waste containers are used to hold radioactive materials as they are being removed from the plant during decommissioning. The radioactive material in/on these containers is material that is already accounted for above when considering the contamination contained on plant surfaces. Thus, radioactive waste containers do not result in significant additional releasable source term beyond that already considered.

The assumed radioactive material on plant surfaces is derived from the results of the LACBWR initial site characterization performed in 1998 following permanent shutdown and decay-corrected to December 2012 (Reference 4.8.4). Specifically, the radioactivity on plant surfaces is conservatively estimated by assuming that the surface contamination present is at levels twice those determined from the LACBWR site characterization. Doubling the site characterization results is intended to provide sufficient margin for the unexpected but potential discovery of localized radiological contamination that could exceed amounts estimated by site characterization measurements. Radioactive decay since December 2012 is ignored in the analysis. Since much of the remaining radionuclide inventory is of relatively long half-life, this assumption ensures reasonably conservative values for the remaining source term.

Using the above-described assumptions, approximately 1.175 Ci of radioactive material is conservatively estimated in the analysis to be present on plant surfaces, and as such represents the assumed total non-ISFSI radioactive source term remaining at the LACBWR site. The LACBWR analysis of postulated release events separately considers the portion of this remaining radioactive contamination that is immediately releasable as airborne contamination and that immediately releasable as contaminated liquid.

4.3.1.1 Portion of Total Radioactivity Assumed Releasable Via the Airborne Pathway

A conservative fraction of 30 percent of the total remaining source term is assumed in the analysis to be immediately available for airborne release. This assumption is reasonably conservative while ensuring that the analysis results well bound the consequences of a postulated airborne release during the LACBWR decommissioning. Specifically, the vast majority of radioactive material remaining at LACBWR is in the form of fixed surface contamination on plant SSCs.[1] The removal and/or decontamination of these SSCs inherently involves the potential generation of airborne radioactive particulates (e.g., grinding, chemical decontamination, or thermal cutting of contaminated components).[2]

However, radioactive contamination is distributed throughout numerous SSCs and over relatively large areas. Industry experience at previously decommissioned nuclear reactor plants demonstrates that dismantlement/decontamination is done in distinct manageable "pieces." For example, a system or several small systems, and/or portions thereof, may be designated for removal and/or decontamination at any one time. After that effort is completed, the next system or systems is addressed. The radioactive material collected during each effort is processed, packaged, and shipped on an ongoing basis, such that its accumulation on site is limited. This "piece-by-piece" process inherently ensures that there is no reasonable likelihood that a significant fraction of the total remaining radioactive material could be simultaneously disturbed and released as airborne particulate.

Based on the above, it is determined that an assumed fraction of 30 percent of the total remaining source term represents a conservative bounding value for the LACBWR post-fuel accident analysis. Additional assumptions used in the analysis of a postulated airborne release event are described in Section 4.3.2 below.

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1 Airborne contamination is minimized by minimizing loose contamination levels and their sources. The use of installed and temporary ventilation systems prevents the build-up of air contamination concentrations.

2 Airborne radioactive particulate emissions will continue to be filtered, as applicable, and effluent discharges monitored and quantified. This includes (1) the operation of appropriate portions of building ventilation systems, or approved alternate systems, as necessary during decontamination and dismantlement activities; and (2) use of local high efficiency particulate air (HEPA) filtration systems for activities expected to result in the generation of airborne radioactive particulates

4.3.1.2 Portion of Total Radioactivity Assumed Releasable Via the Liquid Pathway

Potential sources of radioactive liquid that remain at LACBWR include water generated during decommissioning/decontamination activities (e.g., draining, decontamination, and cutting processes). The portion of the total remaining source term conservatively assumed in the analysis to be available for liquid release at any one time is radioactively contaminated liquid of the following volume, radionuclide concentration, and release flow rate associated with the retention tank contents:

1. 80 percent of the total 6000 gallon volume of the retention tank, which is 4800 gallons.
2. Maximum total radionuclide concentration of 3.9E-03 μCi/cc, which based on the LACBWR-specific radionuclide mix corresponds to a Co-60 concentration of 3.6E-03 μCi/cc.
3. Maximum flow rate from the retention tank of 20 gpm.

This assumption is reasonably conservative while ensuring that the analysis results well bound the consequences of a postulated liquid release during the LACBWR decommissioning. Specifically, the selection of "80 percent" of the total tank volume is an NRC-accepted conservative assumption, based on the Staff guidance of Branch Technical Position (BTP) 11-6, as further clarified in DC/COL-ISG-013. The assumption that the total radionuclide concentration of the retention tank contents is less than or equal to 3.9E-03 μCi/cc is also conservatively bounding. The value of 3.9E-03 μCi/cc is sufficiently above minimum detectable levels for the monitoring instrumentation used at LACBWR, while also allowing for operational flexibility considering the radionuclide concentrations anticipated to be generated by decommissioning activities.

The vast majority of radioactive material remaining at LACBWR is in the form of fixed surface contamination on plant SSCs. The removal and/or decontamination of these SSCs inherently involves the potential generation of liquid radioactive waste (e.g., as a result of draining, decontamination, and cutting processes during plant decommissioning). The "piece-by-piece" decommissioning process discussed in Section 4.3.1.1 above inherently ensures that there is no reasonable likelihood that a significant fraction of the total remaining radioactive material could be released as radioactively contaminated liquid. Any contaminated liquids that are generated during decommissioning are contained within existing or supplemental barriers and processed (i.e., recirculated, sampled, and diluted) to ensure the radionuclide concentration of the retention tank contents does not exceed an appropriate operational limit established in LACBWR procedures. This operational limit incorporates sufficient margin to the 3.9E-03 μCi/cc limit to ensure that, with allowance for instrumentation uncertainty, the design-basis 3.9E-03 μCi/cc limit will not be exceeded.

Finally, the post-fuel accident analysis demonstrates that, in the unlikely event that 80 percent of the retention tank volume at a total radionuclide concentration of 3.9E-03 μCi/cc were to be released from the retention tank at a flow rate of 20 gpm, the normal effluent concentration limits of 10 CFR 20, Appendix B, Table 2, would not be exceeded (see Section 4.4). Thus, the 20 gpm maximum flow rate from the retention tank is a reasonable value to be established as a design-basis limit. An appropriate operational limit is established in LACBWR procedures that incorporates sufficient margin to the 20 gpm limit. This margin ensures that, with allowance for instrumentation uncertainty, the design-basis 20 gpm limit will not be exceeded.

Based on the justification documented above, this assumption represents a reasonably conservative bounding input to the analysis. Additional assumptions used in the analysis of a postulated liquid release event are described in Section 4.3.3 below.

4.3.2 Additional Assumptions – Postulated Airborne Release

The following assumptions were used in the LACBWR analysis of a postulated airborne release scenario:

4.3.2.1 Genoa-3 (G-3) Office Building Occupancy

For the LACBWR post-fuel accident analysis, it is assumed that an individual working in the G-3 office building stays in the building for 10 hours. This is reasonably conservative since it exceeds by two hours the typical work day duration of 8 hours.

4.3.2.2 Terrain Height Above Grade

The X/Q methodology of Regulatory Guide 1.145 (Reference 4.8.7)] uses the terrain height above grade to calculate the effective stack height. The terrain height difference over the LACBWR site is negligible. Therefore, for purposes of the post-fuel accident analysis, it is assumed that the terrain height is the same as plant grade.

4.3.2.3 σ_Y and σ_Z at Distances Less Than 100 Meters

The NRC regulatory guidance governing development of σ_Y and σ_Z do not provide σ_Y and σ_Z values at distances less than 100 meters. Thus for the LACBWR post-fuel accident analysis, the methodology used to obtain σ_Y and σ_Z at distances less than 100 meters is derived from the equations and figures in "Meteorology and Atomic Energy" (M&AE – Reference 4.8.5) and linearly extrapolated to distances less than 100 meters. It is assumed that the σ_Y and σ_Z values used in the analysis at 60 m and 70 m are reasonably representative because they are extrapolated from a region of the curve that is essentially linear.

4.3.2.4 Pasquill Stability Class

It is conservatively assumed that the meteorological category is Pasquill Stability Class F.

4.3.2.5 Rem vs Rad

For the purposes of the LACBWR post-fuel accident analysis, 1 rad is assumed to be equivalent to 1 rem. This is acceptable because the calculated exposures (in rad) are a small fraction of the total dose.

4.3.2.6 Thyroid Dose

The dose to the thyroid is not considered in determining if the dose criteria are met. This has no significant effect on the analysis results since:

1. There is no radioiodine present in the LACBWR site (non-ISFSI) radionuclide inventory; and
2. The CEDE dose conversion factor (DCF) for the only other thyroid significant nuclide, Co-60, is approximately 3.5 times greater than the thyroid DCF. Since the CEDE DCF is

larger, and the CEDE acceptance criterion is lower, the limiting dose is the CEDE dose rather than the thyroid dose.

4.3.2.7 Correction Factor (CF) for G-3 Office Building

Radioactivity inside the G-3 office building is a function of time. The analysis considers two time periods:

1. 0 to1800 seconds – Radioactivity builds up over the first 30 minutes when the fumigation X/Q is used. During this period the inlet concentration is determined by the fumigation X/Q.

2. 1800 to 36,000 seconds – Radioactivity is exhausted over the remaining 9-½ hours when the non-fumigation X/Q is used. During this period the inlet concentration is 0.0 because the elevated release X/Q is 0.0.

4.3.2.8 Radionuclide Data

For the accident doses and doses for alpha emitting radionuclides (Pu, Am, Cm), the dose conversion factors were taken from Federal Guidance Report No. 11 (Reference 4.8.6). Doses are early phase projections during the first two hours or less.

4.3.2.9 Atmospheric Release Inputs

The following values were used in the analysis.

Input Parameter	Value
Distance; Release Point to Road	50 m
Distance; Release Point to G-3 Office Building	70 m
Distance; Release Point to Front Gate	120 m
Stack Height, h_s	350 ft-0 in
Breathing Rate	3.47E-04 m^3/sec
Fumigation Condition Duration	One-half hour
Elevated Wind Speed, U_{he}	2 m/sec
Release Duration	2 hours

4.3.3 Additional Assumptions – Postulated Liquid Release

The following assumptions were used in the LACBWR analysis of a postulated contaminated liquid release:

4.3.3.1 Retention Tank Release, Dilution, and Mixing

It is assumed that the release is fully diluted and mixed at the Thief Slough outlet (which empties into the Mississippi River). This is a reasonable location for the analyses because the nearest drinking water intake is 195 miles downstream and the Thief Slough outlet is the nearest sport fishing location. Also, the river is not used for irrigation, and the shoreline deposits pathway is insignificant for the Mississippi River. It is reasonable to assume complete mixing because the transit time to the Thief Slough outlet is 1.1 hours, and the annual average dilution factor is 107.

4.3.3.2 Duration of Retention Tank Rupture Release to Thief Slough

The 6000-gallon retention tank is below grade in the containment building. For the postulated non-mechanistic tank rupture scenario, it is conservatively assumed that contaminated water enters the slough at a uniform rate over 24 hours. This would require the tank water to leak out of the containment building and travel underground from the containment building to the slough at 3.33 gpm. There are no credible phenomena that could reasonably be postulated to cause such a release.

4.3.3.3 Thief Slough and G-3 Outfall Flow

The G-3 Outfall Circulating Water flow is withdrawn from and returned to Thief Slough. Reflecting this configuration, it is assumed that the G-3 Outfall Circulating Water flow has no net effect on total flow in or out of the slough.

4.3.3.4 Liquid Release Inputs

The following values were used in the analysis.

Input Parameter	Value
Retention Tank Volume	6000 gal
Minimum Mississippi River Flow	2250 cfs
Conversion from gal to cc	3785.4 ml/gal
Minimum G-3 Circulating Water Flow	43,840 gpm
Fraction of Flow Through Thief Slough	25 percent
Annual Average Dilution Factor for Thief Slough	107

4.4 SUMMARY OF ANALYSIS RESULTS

4.4.1 Postulated Airborne Release

The results of the LACBWR post-fuel accident analysis involving a postulated airborne release are summarized in Table 4-1. As indicated in Table 4-1, the following four doses are calculated:

1. The dose to a person at the edge of the access road;
2. The dose to a person located in the G-3 parking lot;
3. The dose to a person working inside the G-3 office building; and
4. The dose to a person at the G-3 entry gate.

The analysis results summarized in Table 4-1 demonstrate that the consequences of releasing 30 percent of the non-ISFSI radioactive source term remaining at the LACBWR site to the atmosphere are well within the applicable 10 CFR 100.11 and EPA PAG limits.

4.4.2 Postulated Liquid Release

The results of the LACBWR post-fuel accident analysis involving a postulated liquid release are summarized in Table 4-2. As indicated in Table 4-2, the following three postulated release scenarios were evaluated:

1. A (non-mechanistic) retention tank rupture with a direct release to Thief Slough;
2. A 20 gpm release rate directly to Thief Slough; and
3. A 20 gpm release rate into the minimum Genoa-3 Circulating Water flow, which empties into Thief Slough.

The analysis results are summarized in Table 4-2. These results demonstrate that the consequences of releasing 4800 gallons of water containing a radionuclide concentration of 3.90E-03 µCi/cc are less than the normal effluent concentration limit (1E-3 µCi/ml) of 10 CFR 20, Appendix B, Table 2, Column 2, for all three liquid release scenarios. It is noted that the release consequences for all three scenarios also are less than the 10 CFR 20.2003 annual release limits for disposal into sanitary sewerage systems. Although the 10 CFR 20.2003 limits are not directly applicable to these scenarios, the fact that the liquid release results are less than those limits further demonstrates the conclusion that the postulated releases would not have an adverse impact on the health and safety of the public or the environment.

4.5 RADIOLOGICAL OCCUPATIONAL SAFETY

Radiological events could occur that result in increased exposure of decommissioning workers to radiation. However, the occurrences of these events are minimized or the consequences are mitigated through the implementation of the LACBWR Radiation Protection Program. The Radiation Protection Program is applied to activities performed onsite involving radioactive materials. A primary objective of the Radiation Protection Program is to protect workers and visitors to the site from radiological hazards during decommissioning. The program requires LACBWR and its contractors to provide sufficient qualified staff, facilities, and equipment to perform decommissioning activities in a radiologically safe manner.

Activities conducted during decommissioning that have the potential for exposure of personnel to either radiation or radioactive materials will be managed by qualified individuals who will implement program requirements in accordance with established procedures. Radiological hazards will be monitored. The Radiation Protection Program at LACBWR implements administrative dose guidelines for TEDE to ensure personnel do not exceed federal 10 CFR 20 dose limits for occupational exposure to ionizing radiation.

LACBWR work control procedures will ensure that work specifications, designs, work packages, and radiation work permits involving potential radiation exposure or handling of radioactive materials incorporate effective radiological controls.

4.6 OFFSITE RADIOLOGICAL EVENTS

Offsite radiological events related to decommissioning activities are limited to those associated with the shipment of radioactive materials. Radioactive shipments will be made in accordance with applicable regulatory requirements. The LACBWR Radiation Protection Program, Process Control Program, and the Decommissioning Quality Plan assure compliance with these requirements such that both the probability of occurrence and the consequences of an offsite event do not significantly affect the public health and safety.

4.7 NON-RADIOLOGICAL EVENTS

Decommissioning LACBWR may require different work activities than were typically conducted during normal plant operations. However, effective application of the LACBWR safety program to decommissioning activities will ensure worker safety. No decommissioning events were identified that would be initiated from non-radiological sources that could significantly impact public health and safety.

Hazardous materials handling will be controlled by the LACBWR Process Control Program and the corporate Hazardous Material Control Program using approved procedures. There are no chemicals stored onsite in quantities which, if released, could significantly threaten public health and safety.

Flammable gases stored onsite include combustible gases used for cutting and welding. Safe storage and use of these gases and other flammable materials is controlled through the Fire Protection Program and plant safety procedures.

Plant safety procedures and off-normal instructions have been established which would be implemented if a non-radiological event occurred at LACBWR. Implementation of these programs and procedures ensures that the probability of occurrence and consequence of onsite non-radiological events do not significantly affect occupational or public health and safety. Plant safety procedures provide personnel safety rules and responsibilities. These safety procedures control both chemical and hazardous waste identification, inventory, handling, storage, use, and disposal.

4.8 REFERENCES

4.8.1 Sargent & Lundy Calculation No. 2013-03098, "Doses from Release of Site Non-ISFSI Radioactivity"

4.8.2 Environmental Protection Agency (EPA) 400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," October 1991

4.8.3 *U.S. Code of Federal Regulations*, "Standards for Protection Against Radiation," Part 20, Chapter I, Title 10, "Energy" (10 CFR 20)

4.8.4 LACBWR Technical Report No. LAC-TR-138, "Initial Site Characterization Survey for SAFSTOR," Revised December 2012

4.8.5 Meteorology and Atomic Energy 1968, Slade, D. H., Editor, TID-24190, July, 1968, http://www.osti.gov/energycitations/product.biblio.jsp?osti_id=4492043

4.8.6 Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA-520/1-88-020, 1988

4.8.7 NRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Rev. 1, February 1983

4.8.8 Accidental Radioactive Contamination of Human Food and Animal Feeds: Recommendations for State and Local Agencies, US Department of Health and Human Services, 08/13/1998

Table 4-1 Summary Results of 2-Hour Airborne Release Analysis

Location	Dose (rem) (*Note 1*)	Acceptance Criteria (rem)	Meets Criterion
Edge of Access Road (50 m)			
CEDE	0.065		
Immersion	<1.0E-04		
TEDE	0.065	1.0 rem TEDE	Yes
Genoa 3 Parking Lot (70 m)			
CEDE	0.046		
Immersion	<1.0E-04		
TEDE	0.046	1.0 rem TEDE	Yes
Genoa 3 Office Building (70 m) (*Note 2*)			
CEDE	0.038		
Immersion	<1.0E-04		
TEDE	0.038	1.0 rem TEDE	Yes
Front Gate (120 m)			
CEDE	0.027		
Immersion	<1.0E-04		
TEDE	0.027	1.0 rem TEDE	Yes

Notes:

1. 1 rem = 1 rad (see Section 4.3.2.5)
2. Dose reflects assumed 10 hour occupancy (see Section 4.3.2.1).

Table 4-2 Summary Results of Liquid Release Analysis

Release Description	Results	Acceptance Criteria	Meets Criterion
Tank Rupture to Thief Slough			
Sum of Fractions	0.02894	Sum ≤ 1.0	Yes
Total Quantity Released (Ci) (*Note 1*)	0.07086	Total < 1.0 Ci	Yes
20 gpm Discharge to Thief Slough			
Sum of Fractions (*Note 2*)	0.1736	Sum ≤ 1.0	Yes
Total Quantity Released (Ci) (*Note 1*)	0.07086	Total < 1.0 Ci	Yes
20 gpm Discharge to G-3 Outfall			
Sum of Fractions			
G-3 Circulating Water	0.9999	Sum ≤ 1.0	Yes
Slough Outlet (*Note 2*)	0.1736	Sum ≤ 1.0	Yes
Total Quantity Released (Ci) (*Note 1*)	0.07086	Total < 1.0 Ci	Yes

Notes:

1. Total radionuclide concentration in the tank is 3.900E-03 Ci/cc and the tank volume is 1.817E+07 cc; thus, the total activity released is (3.900E-03 Ci/cc x 1.817E+07 cc x 1.0E-06 Ci/Ci =) 0.07086 Ci.
2. The G-3 Outfall Circulating Water flow affects the sum of the fractions only at the outfall, not at the outlet of Thief Slough. Thus, the sum of the fractions for a 20 gpm release rate is 0.1736 at the slough outlet regardless of the G-3 Outfall Circulating Water flow.

5.0 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 decommissioning and dismantlement 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 decommissioning and dismantlement activities is defined in the "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities (GEIS)," NUREG-0586, Supplement 1, November 2002. For decommissioning, the NRC uses a standard of significance derived from the Council on Environmental Quality (CEQ) terminology. The NRC has defined three significance levels: SMALL, MODERATE, and LARGE:

SMALL – Environmental impacts are not detectable or are so minor that they will neither destabilize nor noticeably alter any important attribute of the resource.

MODERATE – Environmental impacts are sufficient to alter noticeably, but not to destabilize, important attributes of the resource.

LARGE – Environmental impacts are clearly noticeable and are sufficient to destabilize important attributes of the resource.

The environmental impact of all completed or planned LACBWR decommissioning and dismantlement activities is SMALL as determined by the GEIS. LACBWR decommissioning is specifically evaluated in the GEIS. As stated in the GEIS, licensees can rely on information in this Supplement as a basis for meeting the requirements in 10 CFR 50.82(a)(6)(ii). 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 LTP for LACBWR will detail final decommissioning activities including site remediation, survey of residual contamination, and determination of site end-use. A final supplement to the Environmental Report in support of the LTP will address all environmental impacts of the license termination stage.