LACROSSE*SOLUTIONS*

10 CFR 50.82(a)(7)
10 CFR 50.71(e)(4)
10 CFR 2.390(b)(4)

June 27, 2016 LC-2016-0019

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

La Crosse Boiling Water Reactor
Facility Operating License No. DPR-45
NRC Docket Nos. 50-409 and 72-046

Subject: Notification of Amended Decommissioning Plan and Post-Shutdown Decommissioning Activities Report (D-Plan/PSDAR) for La Crosse Boiling Water Reactor (LACBWR)

References:

1) Barbara Nick, Dairyland Power Cooperative, Letter to U.S. Nuclear Regulatory Commission, "Application for Order Approving License Transfer and Conforming Administrative License Amendments," dated October 8, 2015
2) Marlayna Vaaler, U.S. Nuclear Regulatory Commission, Letter to Barbara Nick, Dairyland Power Cooperative, "Order Approving Transfer of License for the La Crosse Boiling Water Reactor from the Dairyland Power Cooperative to LaCrosse*Solutions*, LLC and Conforming Administrative License Amendment" dated May 20, 2016
3) William Berg, Dairyland Power Cooperative, Letter to U.S. Nuclear Regulatory Commission, "LACBWR Decommissioning Plan and Post-Shutdown Decommissioning Activities Report revision-March 2014," dated March 12, 2014

LaCrosse*Solutions*, LLC (LS) and Dairyland Power Cooperative (DPC) submitted an Application for License Transfer and Conforming Administrative License Amendments for La Crosse Boiling Water Reactor (LACBWR) to the Nuclear Regulatory Commission (NRC) for review in a letter dated October 8, 2015 (Reference 1). LS is submitting the enclosed amended D-Plan/PSDAR outlining the plan of activities following approval of the license transfer as documented in Reference 2.

This submittal is provided to notify the NRC of a significant schedule change to the PSDAR in accordance with 10 CFR 50.82(a)(7), by which we intend to accelerate the decommissioning schedule following license transfer approval. Decommissioning cost changes reflecting the current decommissioning strategy are also provided.

NMSS01
NMSS26

The LACBWR Decommissioning Plan / Post Shutdown Decommissioning Activities Report (D-Plan/PSDAR) is also recognized as the site Safety Analysis Report (SAR) equivalent as it contains the plant post-fuel accident analysis. Therefore it is subject to revisions being submitted to the NRC on a 24 month interval in accordance with 10 CFR 50.71(e)(4). The last change to the D-Plan/PSDAR occurred in March 2014 as documented in Reference 3. It was agreed with the NRC staff to postpone the biennial update to the D-Plan until the license transfer was implemented.

Attachment 1 provides the June 2016 revision of the D-Plan/ PSDAR. It is a major rewrite to reflect the license transfer to LS and revised decommissioning strategy presented in the proposed License Termination Plan. As such, it is submitted as a complete replacement and is not accompanied by a list of effective pages, page change instructions, or a summary of changes. Revision bars are provided to identify changes.

This submittal contains a Solutions Proprietary Financial Information Affidavit pursuant to 10 CFR 2.390. The Affidavit sets forth the basis for which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in 10 CFR 2.390(b)(4). All documents within the scope of this affidavit are marked as "withhold from public disclosure under 10 CFR 2.390."

Section 3 of the D-Plan/PSDAR contains the proprietary financial information Solutions is providing to the NRC and seeks to have withheld from public disclosure in its entirety. Attachment 2 of this submittal contains a redacted version of D-Plan/PSDAR Section 3 for public disclosure.

As Vice President- Regulatory Affairs, I certify that the information in this submittal accurately presents changes made since the previous submittal and reflects information and analysis performed under the provisions of 10 CFR 50.59 and 10 CFR 50.82 but not previously submitted to the NRC.

There are no regulatory commitments contained in this submittal. If you have any questions regarding this letter, please contact Jim Ashley at (224) 789-4017.

Respectfully,

Gerard van Noordennen

Gerard van Noordennen
Vice President Regulatory Affairs

Attachments:

1) Attachment 1, LACBWR D-Plan/PSDAR, June 2016
2) Attachment 2, LACBWR D-Plan/PSDAR Section 3 Redacted Version

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LaCrosse*Solutions*, LLC PROPRIETARY FINANCIAL INFORMATION AFFIDAVIT

Affidavit of Gerard van Noordennen, Vice President Regulatory Affairs, LaCrosse*Solutions*, LLC.

D-Plan/PSDAR Section 3, contained in Attachment 1 of this submittal, consists of proprietary financial information that LaCrosse*Solutions*, LLC considers confidential. Release of this information would cause irreparable harm to the competitive position of LaCrosse*Solutions,* LLC. The basis for this declaration is:

i. This information is owned and maintained as proprietary by LaCrosse*Solutions*, LLC,

ii. This information is routinely held in confidence by LaCrosse*Solutions*, LLC and not disclosed to the public,

iii. This information is being requested to be held in confidence by the NRC by this petition,

iv. This information is not available in public sources,

v. This information would cause substantial harm to LaCrosse*Solutions*, LLC if it were released publicly, and

vi. The information to be withheld is being transmitted to NRC in confidence.

I, Gerard van Noordennen, being duly sworn, state that I am the person who subscribes my name the foregoing statement, I am authorized to execute the Affidavit on behalf of LaCrosse*Solutions*, LLC, and that the matters and facts set forth in the statement are true to the best of my knowledge, information, and belief.

Gerard van Noordennen

Name: Gerard van Noordennen
Title: Vice President Regulatory Affairs
Company: LaCrosse*Solutions*, LLC

SUBSCRIBED AND SWORN TO BEFORE ME

THIS 27th DAY of June, 2016

Notary Public

LINDA CHOU
Official Seal
Notary Public - State of Illinois
My Commission Expires Dec 7, 2019

Attachment 1

LACBWR

D-Plan/PSDAR

June 2016

LA CROSSE BOILING WATER REACTOR

DECOMMISSIONING PLAN

AND

POST-SHUTDOWN DECOMMISSIONING ACTIVITIES REPORT

REVISED

June 2016

PREPARED BY _______________ DATE 6-2-16

REVIEWED BY _______________ DATE 6-2-16
Qualified Technical Reviewer, Regulatory Affairs

APPROVED BY _______________ DATE 6/6/2016
General Manager

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1.0 DESCRIPTION OF PLANNED DECOMMISSIONING ACTIVITIES

1.1. INTRODUCTION

The La Crosse Boiling Water Reactor (LACBWR) was a 50 Megawatt Electric (MWe) BWR that is owned by Dairyland Power Cooperative (Dairyland). This unit, also known as Genoa 2, is located on the Dairyland Genoa site on the east shore of the Mississippi River south of the Village of Genoa, Vernon County, Wisconsin.

The site is licensed under Possession Only License No. DPR-45 with Docket Numbers of 50-409 for LACBWR and 72-046 for the Independent Spent Fuel Storage Installation (ISFSI). LACBWR has been shut down since 1987 and is currently undergoing decommissioning. The spent nuclear fuel stored in the LACBWR ISFSI will be maintained under an amended Part 50 license.

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.

1.1.1. Historical Selection Of Decommissioning Method

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.

Following plant shutdown, 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 at the time of shutdown 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.

1.1.2. Current Selection Of Decommissioning Method

In a letter dated October 8, 2015 (Reference 1), Dairyland and LaCrosse*Solutions*, LLC (Solutions) requested Nuclear Regulatory Commission (NRC) consent to transfer Dairyland's possession, maintenance and decommissioning authorities, under Possession Only License No. DPR-45, from Dairyland to Solutions. Following NRC approval, as documented in a letter dated May 20, 2016 (Reference 2), the LACBWR site is being decontaminated and dismantled in accordance with the DECON alternative.

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 (Reference 3), and a possess but not operate status was granted. The Decommissioning Plan was submitted December 1987 (Reference 4) with a chosen decommissioning alternative of SAFSTOR. License Amendment No. 63, dated August 18, 1988 (Reference 5), 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 (Reference 6). 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 (Reference 7), 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 (Reference 8).

License Amendment No. 69, containing the SAFSTOR Technical Specifications, was issued April 11, 1997 (Reference 9). 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 (Reference10), 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. All spent fuel assemblies and fuel debris were placed in dry cask storage in the ISFSI in September 2012.

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.

License Amendment No. 72 was issued July 31, 2013 (Reference 11), 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 Fuel Element Storage Well.

On May 20, 2016, The NRC approved the transfer of Dairyland's possession, maintenance and decommissioning authorities, under Possession Only License No. DPR-45, from Dairyland to Solutions (Reference 2).

In a letter dated March 31, 2016, the NRC withdrew the September 15, 1994 Confirmatory Order modifying the NRC Order Authorizing Decommissioning of facility for LACBWR (Reference 14)

The License Termination Plan (LTP) for LACBWR (Reference 12) was submitted for NRC review in June 2016. It details final decommissioning and dismantlement activities including site remediation and survey of residual contamination.

The June 2016 revision of the Decommissioning Plan and Post-Shutdown Activities Report (DP/PSDAR) identifies DECON activities, a revised decommissioning schedule, and an updated cost estimate provided by Solutions.

1.3. DISMANTLEMENT OF SYSTEMS AND COMPONENTS

Significant dismantlement has already been accomplished. In excess of 2 million pounds of metallic waste have been removed, shipped, and disposed of in addition to the Reactor Pressure Vessel (RPV) and spent fuel storage racks. Removal and disposal of the RPV included disposition of irradiated hardware and all other Class B and C waste.

Waste stored in the Fuel Element Storage Well (FESW) was processed and collected with other Class B/C waste (i.e., resins, filters, and waste barrel contents) and packaged in three liners that were shipped for disposal in June 2007. The RPV containing the reactor internals and 29 control rod blades was filled with low-density cellular concrete with the reactor head installed. Attachments to the RPV were removed and all other appurtenances were cut. The RPV was removed from the Reactor Building and was shipped for disposal in June 2007. After all spent fuel assemblies and fuel debris were placed in dry cask storage in the ISFSI in September 2012, the storage racks and installed components were removed from the FESW.

The remaining systems being decontaminated and dismantled by Solutions in accordance with the DECON method are described below.

1.3.1. Forced Circulation System

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. This system is drained and approximately 90% removed.

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. This system is drained.

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. Ion exchange resins have been removed and the system has been drained. The Primary Purification pump, coolers and 90% of the piping have been removed and disposed as radioactive waste.

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. Approximately 50% of the system in the Reactor Building was removed due to interference with the reactor vessel removal and 80% of the piping in the Turbine Building has been removed. Remaining system components are drained and are ready for dismantlement.

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. This system has been removed with the exception of the underground gas storage tanks and the gas storage tank in the Turbine Building tunnel.

1.3.7. Fuel Element Storage Well System

The Fuel Element Storage Well (FESW) is a stainless steel lined concrete structure that measures 11 feet by 11 feet by approximately 42 feet deep. When full, it contained approximately 38,000 gallons of water. The FESW cooling system is connected to the well and consists of two pumps, one heat exchanger, one ion exchanger, piping, valves, and instrumentation. All spent fuel and fuel debris, installed components, and storage racks have been removed from the storage well and a fixative was applied to the walls. The system has been drained.

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 secondary cooling systems. The system provided cooling water to Reactor Building Air Conditioner compressors. The dismantlement of this system is approximately 70% complete.

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. The hydraulic system has been drained. The air compressors, water pumps, and other equipment have also been electrically disconnected and drained.

1.3.10. Well Water System

Water for this system was supplied from two sealed submersible deep well pumps that took suction through stainless steel strainers, and discharged into integrated pressure tanks. The system supplied water to the plant and office for sanitary and drinking purposes. It was used as cooling water for the two Turbine Building air-conditioning units and the heating boiler blow-down flash tank and sample cooler. The well water system also supplied water for laundry equipment and seal water for the Circulating Water pumps. The Well Water System has been isolated from the Reactor/Turbine building. Well water pump #3 supplies potable water to the Administration Building and Crib House. This system is maintained in continuous operation.

1.3.11. Demineralized Water System

The Demineralized Water System consists of a lower Condensate Storage Tank with a capacity of 19,100 gallons and an upper Virgin Water Tank, which has a capacity of 29,780 gallons. Both are two sections of an integral aluminum tank located on the Turbine Building office roof. The Virgin Water Tank provided water to the Demineralized Water transfer pumps, which distributed demineralized water throughout the plant. Water was demineralized in batches at Genoa Fossil Station (G-3). The system 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 FESW, cask pool, and upper cavity during cask loading operations. During operations, 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. The system is currently drained and inoperable.

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. This system is currently still operable and is used as needed.

1.3.14. Low Pressure Service Water System

Two vertical pumps located in the LACBWR Crib House supplied the system through a duplex strainer unit. The Low Pressure Service Water (LPSW) system supplied the CCW coolers and was the normal supply to the HPSW system through the motor-driven HPSW pump. This system is currently used as the dilution water supply for conducting liquid waste discharges of river water ingress in the turbine building tunnel area. The original system has been simplified to allow dismantlement which is approximately 30% complete. The system is supplied by one vertical pump located in the Cribhouse through a duplex strainer unit. This system is maintained available for periodic dilution of liquid waste discharges.

1.3.15. High Pressure Service Water System

The purpose of the HPSW system is to supply fire suppression water. The adjacent coal plant (G-3) maintains HPSW system pressure. The HPSW system is divided into two main loops. The internal loop served the Turbine Building, Reactor Building, and Waste Treatment Building (WTB) interior hose stations and sprinkler systems. The external loop supplies outside fire hydrants and the LACBWR Crib House sprinklers. The external loop was cross-connected with the fire suppression system for G-3. The system is isolated from the Turbine Building by the removal of isolation valves and installing pipe blanks to allow only external fire loop use. This system is maintained operational for fire protection.

1.3.16. Circulating Water System

Circulating water is drawn into the intake flume from the river by two pumps located in separate open suction bays in the LACBWR Crib House. 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 LACBWR Crib House. The Circulating Water System is inoperable as the system pumps have been de-energized.

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. With the exception of the Condensate Storage Tank (CST), this system has been drained and most of the piping has been removed. It is not maintained operational.

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. This system, with exception of the drained clean and dirty oil tanks, has been completely removed.

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

The Reactor Building Ventilation System utilized two 30 ton, 12,000 cfm air conditioning units for drawing fresh air into the building and for circulating the air throughout the building. Each air conditioning unit air inlet included a filter box assembly, face and bypass dampers, and one 337,500 Btu/hr capacity steam coil that was used when heating was required. Air entered the building through openings between and around the bi-parting door sections and was exhausted

from the building using stack blowers through the exhaust stack for the facility. The stack is 350 feet high and is constructed of structural concrete with an aluminum nozzle at the top. The exhaust system was equipped with conventional and high-efficiency filters and monitored by a particulate radiation detector. A monitoring system is installed with the exhaust system, which includes an isokinetic nozzle, located downstream of the high-efficiency filters. The system, which historically has drawn air from the stack exhaust to the monitoring system, is capable of detecting particulate and gaseous activity.

Ventilation for the WTB was provided by a 2000 cfm exhaust fan that drew air from the shielded vault areas of the building and exhausted the air through a duct out the floor of the building to the gas storage tank vault. The stack blowers then exhausted the air from the gas storage tank vault through the connecting tunnel and discharged the air from the building through the stack. Currently, both systems are de-energized. Building ventilation is currently supplied through the use of stack fans that are energized by temporary power.

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. Select HVAC systems are maintained operational.

1.3.20. <u>Liquid Waste Collection Systems</u>

The Liquid Waste System collects the liquid waste from the Turbine Building, the WTB, the gas storage tank vault, and the tunnel into 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, which is separate from the 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 and from the retention tanks.

Following processing and sampling to demonstrate compliance with the release criteria, the liquid waste from each system was then discharged to the common G-3/LACBWR circulating water discharge outflow. This was the normal liquid effluent release pathway for LACBWR.

Spent resin was transferred to the spent resin receiving tank and held until there was a sufficient quantity available for shipment to an approved processing facility. The resin was then transferred to an approved shipping container, dewatered, packaged and shipped for off-site disposal.

Currently, the liquid waste collection system pumps are not operable. The only liquid effluent discharges at LACBWR are through the East Turbine building sump using an air operated pump. The air operated pump discharges to the inlet of the liquid waste monitor, which is still operable.

1.3.21. Electrical Distribution System

The plant is disconnected from off-site power in support of Cold and Dark status. Currently the site has a 480 volt 500 kVa transformer providing power to multiple power carts throughout the buildings. Temporary power is tied to existing overhead cranes and power to exhaust fans for the stack.

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.

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.

1.3.23. Full-Flow Condensate Demineralizer System

The Full-Flow Condensate Demineralizer System consisted of resin-filled service tanks that removed ionic impurities from the condensate water going back to the reactor. All tanks and piping have been removed.

1.3.24. Steam Turbine

The turbine was a high pressure, condensing, reaction, tandem compound, reheat 3600 rpm unit rated at 60 MW. The turbine consisted of a high pressure and intermediate pressure and a low-pressure element. The Steam Turbine, as well as some steam system piping and components have been removed.

1.3.25. 60-Megawatt Generator

The 60 MW generator was a high-speed, turbine-driven wound-rotor machine rated at 76,800 kVA. The generator was cooled by a hydrogen system, lubricated by a forced-flow lubricating system, and excited by a separate exciter attached to the end of the generator shaft through a reduction gear. The main and reserve exciters have been removed. The generator has been completely removed, unconditionally released and dispositioned.

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

Solutions plans for the decontamination, dismantlement and anticipated end-state condition(s) for the remaining site structures are presented in the following sections. The methods to remediate contaminated structures, systems, and equipment do not involve any unique safety or remediation issues.

1.4.1. Reactor Building

The Reactor Building is a right circular cylinder building 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 feet 6 inches below grade level. The steel 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 RPV and biological shielding. The interior of the shell is lined with a 9-inch-thick layer of concrete to an elevation of 727 feet 10 inches Above Mean Sea Level (AMSL). The structure is supported on a foundation consisting of concrete-steel piles and a pile capping of concrete approximately 3 feet thick.

The shell includes two airlocks. The personnel airlock connects the Reactor Building to the Turbine Building. In addition, there is also an 8 feet by 10 feet freight door opening in the Reactor Building that is intended to accommodate large pieces of equipment. To facilitate RPV removal and dry cask storage, an opening was created in the Reactor Building. The opening is closed by a weather tight, insulated, roll-up, bi-parting door. The majority of pipe penetrations leaves the Reactor Building 1 to 10 feet below grade level either at the northwest quadrant or at the northeast quadrant and enters the pipe tunnel.

A 50 ton traveling polar 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. 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 remaining components and systems in the Reactor Building will be drained, dismantled, and removed. Area preparation and set-up for commodity removal will include radiological surveys and the identification and mitigation of any hazardous material.

Systems or components will be removed utilizing mechanical means with support from the overhead crane or local hoists. Some systems or components might require hot work activities to size reduce. The systems will be either loaded into shipping containers or will be transported to the waste loading area in the Turbine Building.

Water and sludge are currently inside the OHST. The tank will be dewatered and desludged. Once the tank has been adequately characterized, a fixative will be applied in preparation for building demolition. Keeping associated piping and components in place while performing this activity will allow for the collection of any liquids. This piping and associated components will be dismantled and removed once the tank is prepared for demolition.

In preparation for the building demolition, the FESW will be dismantled from the main floor elevation. As necessary, prior to dismantlement, the FESW will be decontaminated to levels that will allow for the performance of hot work activities required to remove the stainless steel FESW liner. Associated equipment (HVAC, etc.) and piping will also be removed. Contaminated sections of the FESW liner will be size reduced. Sized pieces of the FESW liner will be lifted by the overhead crane, packaged and placed in the waste staging area in the Turbine Building. Once the FESW liner has been removed, the concrete underlying the FESW will be decontaminated to the extent necessary to allow open-air demolition.

After commodity removal is complete, the basement elevation (621 foot) of the Reactor Building will be remediated to levels that will allow demolition of above ground structures in open air with proper contamination controls in place. The preparation of the Reactor Building basement will include:

- The removal of any identified Hazardous Waste.
- Asbestos Abatement (to be performed by a qualified Subcontractor).
- Dismantle and remove the 6,000-gallon retention tanks.
- Dismantle and remove the two Forced Circulation Pumps, including the inlet and outlet piping as well as the auxiliary oil pumps.
- Dismantle and remove the condensate purification ion exchangers.
- Remediation of the sump.
- The removal of any lead shielding.

The exposed concrete will then be surveyed to determine the scope of the remediation necessary to meet the release criteria specified in the LACBWR License Termination Plan (LTP) (Reference 12). If unacceptable contamination is identified within the first couple of inches of concrete, a scabbling method or air hammering method will be used to remove the contaminated material. If the contamination is detected at a deeper depth, then a wire saw or other means will be used to remove contaminated areas above the limits for unrestricted release.

Once surveys have demonstrated that the open-air demolition criteria have been met, a high reach excavator equipped with a pneumatic hammer will be used to break the interior concrete walls down to grade level. After the above grade internal steel shell is exposed, it will be segmented and removed using cutting torches from the top down to the ground elevation. The remaining interior concrete will then be removed to the thermal shield. The thermal shield will be segmented using a wire saw into manageable pieces for special packaging as mixed waste. After the thermal shield is removed, the remainder of the interior concrete will be removed, completely exposing the steel liner. Subsequent to interior concrete removal, the remaining portion of the steel liner will be removed. Demolition debris will be loaded into gondola railcars for disposal.

The remaining structural concrete outside the liner below the 636 foot elevation (i.e., concrete "bowl" below 636 foot elevation, concrete pile cap and piles) will remain and be subjected to a Final Radiation Survey (FRS) in accordance with LTP Chapter 5.

All structural decontamination activities will be performed in accordance with approved Radiation Work Permits (RWP) and under the oversight of Radiation Protection personnel.

1.4.2. Turbine Building and Turbine Office Building

The Turbine Building housed the steam turbine and generator, main condenser, electrical switchgear, and other pneumatic, mechanical and hydraulic systems and equipment. A 30-ton traveling bridge crane with a 5 ton auxiliary hoist capacity 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 105 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.

Commodity removal has begun and will continue in the Turbine Building utilizing cutting tools and mechanical means to dismantle radioactive piping and components. Saws and torches will be the primary methods used. System pieces and waste will be sized to meet packaging and waste disposal criteria. Waste will be packaged and stored in the waste loading area located on the ground floor of the Turbine Building until it is properly shipped and disposed of at an off-site facility.

After radioactive commodity removal is complete, surveys will be performed of all remaining systems, components and structural surfaces to ensure that the open-air demolition criteria is met. Structural materials that do not meet the open-air demolition criteria will be removed, segregated, properly packaged and disposed of as radioactive waste. As with the Reactor Containment, if unacceptable contamination is identified within the first couple of inches of concrete, a scabbling method or air hammering method will be used to remove the contaminated material. If the contamination is detected at a deeper depth, then a wire saw or air hammer will be used to remove contaminated areas. All radioactive waste will be loaded and transported to an acceptable radioactive waste disposal facility, primarily the EnergySolutions facility in Clive, Utah. All structural decontamination activities will be performed in accordance with approved Radiation Work Permits (RWP) and under the oversight of Radiation Protection personnel.

When all structural surfaces have been decontaminated to the open-air demolition limits, the building will be demolished. All remaining systems and components, interior structural surfaces and structural concrete will be demolished to the ground level slab. Concrete will be processed to meet disposal site requirements. Misting methods will be utilized during building demolition to minimize dust. The majority of the construction debris resulting from the demolition of the structure is expected to be treated as low level radioactive waste and will be shipped to the licensed radioactive waste disposal facility, primarily by gondola railcar. If possible, some materials, including structural concrete will be radiologically surveyed and released for unrestricted use. Materials released for unrestricted use can be recycled or released for disposal at a non-radiological landfill. All bulk material sent for recycle or disposal at a non-radioactive disposal site will be subjected to a final assessment for the presence of any residual radioactive contamination by passing through a radiological truck monitor.

When the Turbine Building has been demolished, the foundation slab will be exposed. An excavator with appropriate tools (pneumatic hammer, loading bucket, etc.) will remove the building slab and foundation to meet disposal site requirements. During slab and foundation removal, underground utilities will also be removed, surveyed, packaged and properly disposed. Some examples of underground utilities might be gas, water, miscellaneous piping, and discharge line. The slab and foundation walls will be removed to a minimum depth of 3 feet below grade (636 foot elevation).

1.4.3. Waste Treatment Building, Gas Storage Tank Vault and LSA Storage Building

The WTB is located to the northeast of the Reactor Building. The building contained facilities and equipment for decontamination and the collection, processing, storage, and disposal of low level solid radioactive waste. 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 foot deep sump with 8 inch thick walls and a bottom which extend to a depth of 626 feet.

The grade floor of the WTB contains a shielded compartment which houses a 320 ft3 stainless steel spent resin receiving tank and resin receiving and transfer equipment. Located outside of the shielded cubicle were two back-washable radioactive liquid waste filters and dewatering piping, containers, and pumps. The main floor of the WTB also housed a decontamination facility, consisting of a steam cleaning booth, a decontamination sink, and heating/ventilation/air conditioning units.

The basement of the WTB consists of two shielded cubicles. One cubicle, to which access is gained by removal of floor shield plugs, was used for the storage of high 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 Gas Storage Tank Vault (GSTV) is a 29 foot by 31 foot underground concrete structure with 14 feet high walls and 2 feet thick floors, walls, and ceiling located below ground just outside of the WTB. The vault is 3 feet below grade with a sump that extends to a depth of 22 feet or elevation 617 feet. The Gas Decay System routed main condenser gases through various components for drying, filtering, recombining, monitoring and holdup for decay. Two 1,600 cubic feet tanks are located in the GSTV. 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 along with some associated piping. The remainder of the Gas Decay system has been removed.

The Low Specific Activity (LSA) Storage Building is located southwest of the Turbine Building. It was used to store processed, packaged and sealed low level dry active waste materials, and sealed low level activity components. No liquid wastes were stored in this building. The LSA Building is 27 feet by 80 feet and 15 feet tall.

Commodity removal has begun and will continue in the WTB, GSTV and LSA Building utilizing cutting tools and mechanical means to dismantle radioactive piping and components. Band saws and reciprocating saws will be the primary methods used. The interiors of tanks will be cleaned and a fixative will be applied prior to dismantlement. System pieces and waste will be sized to meet packaging and waste disposal criteria. Waste will be packaged and stored in the waste loading area in the Turbine Building until it is properly shipped and disposed of at an off-site facility.

After commodity removal is complete, the interior concrete surfaces will be remediated to the open air demolition criteria. Structural material that does not meet the open-air demolition criteria will be removed, segregated, packaged and disposed of as radioactive waste. If unacceptable contamination is identified within the first couple of inches of concrete, a scabbling method or air hammering method will be used to remove the contaminated material. If the contamination is detected at a deeper depth, then a wire saw or air hammering will be used to remove contaminated areas. All radioactive waste will be loaded and transported to an acceptable radioactive waste disposal facility.

When all structural surfaces have been decontaminated to the open-air demolition criteria, the above grade portions of each building will be demolished. The entire LSA building, including the concrete floor slab will be removed. The remaining structural concrete for the WTB and GSTV located below 636 foot elevation will remain and be subjected to a FRS in accordance with LTP Chapter 5. Concrete will be processed to meet disposal site requirements. Misting methods will be utilized during building demolition to minimize dust. All construction debris resulting from the demolition of each of the structure will be treated as low-level radioactive waste and will be shipped to the licensed radioactive waste disposal facility, primarily by gondola railcar.

All structural decontamination activities will be performed in accordance with approved Radiation Work Permits (RWP) and under the oversight of Radiation Protection personnel.

1.4.4. Cribhouse

The LACBWR Crib House is located on the bank of the Mississippi River to the west of the plant. The structure served as the intake for the Circulating Water System, which provided cooling water to various LACBWR plant systems. The LACBWR Crib House contains the diesel-driven high pressure service water pumps, low pressure service water pumps and the circulating water pumps. The LACBWR Crib House is 35 feet by 45 feet and 15 feet tall.

In addition to the G-3 Crib House, the LACBWR Crib House also currently serves the active G-3 facility, and consequently, will remain intact and undisturbed by the decommissioning process. Radiological surveys will be performed on the interior and exterior structural surface and systems of the LACBWR Crib House to demonstrate that the structure can be released for unrestricted use.

1.4.5. Maintenance Eat Shack and 1B Diesel Generator Building

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

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 31 feet by 38 feet and 13 feet tall.

These two structures will be radiologically characterized and then completely demolished, including the concrete foundation slabs. Construction debris resulting from the demolition of these structures will be treated as low-level radioactive waste and will be shipped to the licensed radioactive waste disposal facility, primarily by gondola railcar. If possible, some materials, including structural concrete will be radiologically surveyed and released for unrestricted use. Materials released for unrestricted use can be recycled or released for disposal at a non-radiological landfill. All bulk material sent for recycle or disposal at a non-radioactive disposal site will be subjected to a final assessment for the presence of any residual radioactive contamination by passing through a radiological truck monitor.

1.4.6. Ventilation Stack

The LACBWR Ventilation Stack is a 350 feet high, tapered, reinforced concrete structure with an outside diameter of 7 feet at the top and 25 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 40 foot square concrete base formed without triangular sides on the southeast and southwest corners.

The Ventilation Stack will be radiologically characterized and then completely demolished. Once characterization is complete, interior concrete surfaces will be remediated to open air demolition criteria and demolished. If remediation is not practicable, then a fixative will be applied to the identified surfaces to mitigate any loose contamination.

Demolition will be performed from the top down. It is anticipated that a traveling scaffold system will be employed to provide access to the exterior of the stack. The stack will be removed down to the lower 40 foot elevation. The bottom 40 feet of the ventilation stack will be removed with excavators and pneumatic breakers. The excavators will saw and break the concrete into manageable sections and drop the sections into the interior of stack. An opening at the bottom of the stack will be made to allow the removal of the concrete debris. The concrete foundation mat will be taken down to 3 feet below grade. Wet methods will be utilized during all demolition to minimize dust. Project personal will monitor and barricade the fall zones in the vicinity of the ventilation stack. All construction debris resulting from the demolition of these structures will be treated as low-level radioactive waste and will be shipped to the licensed radioactive waste disposal facility, primarily by gondola railcar.

1.5. RADIOLOGICAL IMPACTS OF DECOMMISSIONING ACTIVITIES

The decommissioning activities described are and will be conducted under the provisions of the Solutions Radiation Protection Program and Radioactive Waste Management Program. The Radiation Protection Program and written site procedures are intended to provide sufficient information to demonstrate that decommissioning activities will be performed in accordance with 10 CFR 19, Notices, Instructions And Reports To Workers, and 10 CFR 20, Standards For Protection Against Radiation and to maintain radiation exposures As Low As Reasonably Achievable (ALARA). The Radioactive Waste Management Program controls the generation, characterization, processing, handling, shipping, and disposal of radioactive waste in accordance with the approved Radiation Protection Program, Process Control Program, and written plant procedures.

The current Radiation Protection Program, Waste Management Program, and Radiological Effluent Monitoring and Offsite Dose Calculation Manual (ODCM) will be used to protect the workers and the public, as applicable, during the various decontamination and decommissioning activities. These well-established programs are routinely inspected by the Nuclear Regulatory Commission (NRC) to ensure that workers, the public, and the environment are protected during facility decommissioning activities.

Continued application of the current and future Radiation Protection and Radiological Effluent Monitoring Programs at LACBWR ensures public protection in accordance with 10 CFR 20 and 10 CFR 50, Appendix I. Radiological Environmental Monitoring Program (REMP) reports for LACBWR to date conclude that the public exposure as a result of decommissioning activities is bounded by the evaluation in NUREG-0586 (Reference 13), which concludes the impact is minimal.

1.5.1. Control Mechanisms to Mitigate the Recontamination of Remediated Areas

Due to the large scope of remaining structures and systems that will be decontaminated and dismantled, FRS of areas may be performed in parallel with decommissioning activities. Consequently, a systematic approach will be employed to ensure that areas are adequately remediated prior to performing FRS and ongoing decommissioning activities do not impact the radiological condition of areas where compliance with the unrestricted release criteria as specified in 10 CFR 20.1402 has been demonstrated. These measures and mechanisms are described in LTP Chapter 5.

1.5.2. Occupational Exposure

The total radiation exposure estimate for remaining decommissioning activities is estimated to be approximately 130.3 person-rem as discussed in LTP Chapter 3.

1.5.3. Exposure to the Public

Continued application of Radiation Protection, Radioactive Waste, Radiological Effluent Technical Specification and Radiological Environmental Monitoring Programs assures public protection in accordance with 10 CFR 20 and 10 CFR 50, Appendix I.

1.5.4. Radioactive Waste Projections

The Radioactive Waste Management Program is used to control the characterization, generation, processing, handling, shipping, and disposal of radioactive waste during decommissioning. Activated and contaminated systems, structures, and components represent the largest volume of low level radioactive waste expected to be generated during decommissioning. Other forms of waste generated during decommissioning include:

- Contaminated water;
- Used disposable protective clothing;
- Expended abrasive and absorbent materials;
- Expended resins and filters;
- Contamination control materials (e.g., strippable coatings, plastic enclosures); and
- Contaminated equipment used in the decommissioning process.

As LACBWR has elected to use an approach commonly referred to as "rip & ship" verses performing significant on-site decontamination activities, the total volume of low-level radioactive waste for disposal has been estimated at 393,696 cubic feet as discussed in LTP Chapter 3. The vast majority of this waste will be shipped to the licensed EnergySolutions radioactive waste disposal facility in Clive, Utah by gondola railcar.

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. This data was also used to support license termination by providing hydrogeological information for 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 support RESRAD, and DCGLs needed during decommissioning actions.

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 and GTCC waste are 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 was isolated from spent fuel pool water during loading in the pool and the transfer cask was 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. ISFSI decommissioning is not within the scope of the Solutions project.

1.8. REFERENCES

1. Letter from Dairyland Power Cooperative to the Nuclear Regulatory Commission, for Order Approving License Transfer and Conforming Administrative License Amendments, dated October 8, 2015.
2. Marlayna Vaaler, U.S. Nuclear Regulatory Commission, Letter to Barbara Nick, Dairyland Power Cooperative, "Order Approving Transfer of License for the La Crosse Boiling Water Reactor from the Dairyland Power Cooperative to LaCrosse*Solutions*, LLC and Conforming Administrative License Amendment" dated May 20, 2016
3. Letter from the Nuclear Regulatory Commission to Dairyand Power Cooperative, Issuance of License Amendment No. 56, dated August 4, 1987
4. Letter from Dairyland Power Cooperative to the Nuclear Regulatory Commission, Submittal of Decommissioning Plan, Preliminary Decon Plan, and Supplement to the Environ Report, dated December 21, 2015.
5. Letter from the Nuclear Regulatory Commission to Dairyand Power Cooperative, Issuance of License Amendment No. 63, dated August 18, 1988
6. Letter from the Nuclear Regulatory Commission to Dairyand Power Cooperative, Order to Authorize Decommissioning and Amendment No. 66 to Possession Only License No. DPR-45 for La Crosse Boiling Water Reactor, dated August 7, 1991
7. Letter from the Nuclear Regulatory Commission to Dairyand Power Cooperative, Confirmatory Order Modifying the August 7, 1991, Decommissioning Order for the La Crosse Boiling Water Reactor, dated September 15, 1994
8. LACBWR Initial Site Characterization Survey for SAFSTOR, dated October 1995

9. Letter from the Nuclear Regulatory Commission to Dairyand Power Cooperative, Issuance of License Amendment No. 69, dated April 11, 1997
10. Letter from the Nuclear Regulatory Commission to Dairyand Power Cooperative, Issuance of License Amendment No. 71, dated January 25, 2011
11. Letter from the Nuclear Regulatory Commission to Dairyand Power Cooperative, Issuance of License Amendment No. 72, dated July 31, 2013
12. La Crosse Boiling Water Reactor License Termination Plan, Revision 0
13. U.S. Nuclear Regulatory Commission NUREG-0586, Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities, Supplement 1, Volume 1 – November 2002.
14. Letter from the Nuclear Regulatory Commission to Dairyand Power Cooperative, Withdrawal of Confirmatory Order, dated March 31, 2016

2.0 SCHEDULE

2.1. SCHEDULE ACTIVITIES COMPLETED PRIOR TO LICENSE TRANSFER TO SOLUTIONS

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. 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.

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 included 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 did not result in significant environmental impacts compared to 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.

2.0 SCHEDULE (cont'd)

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.

2.2. REVISED SOLUTIONS SCHEDULE

The revised schedule for decommissioning activities at LACBWR, reflecting the transfer of decommissioning responsibilities to LaCrosse*Solutions* (Solutions), is depicted in Table 2.1.

Table 2.1 Decommissioning Schedule Milestones

Date	**Milestone**
Q2/2016	License Transfer Complete
Q2/2016	Submit LTP to NRC
Q2/2016	Mobilization Complete
Q3/2016	Stack Demolition Complete
Q3/2017	LTP Approval by NRC
Q3/2017	Component Removal Complete
Q4/2017	Building Demolition Complete
Q4/2018	Transportation and Disposal Complete
Q4/2018	Site Remediation Complete
Q4/2018	FRS Complete
Q1/2019	Site Restoration Complete
Q1/2019	Submit Remaining FRS Reports
Q1/2019	Submit License Transfer to Dairyland Amendment Request to NRC
Q1/2020	License Transfer to Dairyland Approved by NRC
Q1/2020	LACBWR License Termination Approval by NRC

Note: Circumstances can change during decommissioning. If it is determined that the decommissioning cannot be completed as outlined in this schedule, an updated schedule will be provided to the NRC.

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. 1The 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.

4.0 PLANT POST-FUEL ACCIDENT ANALYSIS (cont'd)

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

1. Sargent & Lundy Calculation No. 2013-03098, "Doses from Release of Site Non-ISFSI Radioactivity"
2. Environmental Protection Agency (EPA) 400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," October 1991
3. *U.S. Code of Federal Regulations*, "Standards for Protection Against Radiation," Part 20, Chapter I, Title 10, "Energy" (10 CFR 20)
4. LACBWR Technical Report No. LAC-TR-138, "Initial Site Characterization Survey for SAFSTOR," Revised December 2012
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
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
7. NRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Rev. 1, February 1983
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:

3. 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.

4. 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 has been prepared and details final decommissioning activities including site remediation, survey of residual contamination, and determination of site end-use. Chapter 8 of the LTP contains a supplement to the Environmental Report that describes any new information or significant environmental change associated with the site specific decommissioning and site closure activities performed at the LACBWR site. The LTP supplement to the Environmental Report concluded the following:

As previously evaluated in the D-Plan/PSDAR, the non-radiological environmental impacts from decommissioning LACBWR are temporary and not significant. The potential issues identified as "site-specific" in NUREG-0586 have been evaluated and there is no significant impact. The potential environmental impacts associated with decommissioning LACBWR have already been predicted in and will be bounded by the previously issued environmental impact assessments (NUREG-0191, NUREG-0586, and D-Plan/PSDAR). Therefore, there are no new or significant environmental changes associated with decommissioning.

Attachment 2

LACBWR

D-Plan/PSDAR

Section 3

Redacted Version

3.0 ESTIMATE OF EXPECTED DECOMMISSIONING COSTS

3.1. INTRODUCTION

The revised decommissioning cost estimate prepared by Solutions evaluates the following cost elements:

1. Cost assumptions used, including contingency factors;
2. Major decommissioning activities and tasks;
3. Unit cost factors;
4. Estimated costs for decontamination and removal of equipment and structures;
5. Estimated costs for waste disposal, including disposal site surcharges;
6. Estimated Final Radiation Survey (FRS) costs; and
7. Estimated total costs.

The cost estimate focuses on the remaining work, including costs of labor, materials, equipment, energy, and services. The cost estimate includes the cost of the planned remediation activities as well as the cost of the transportation and disposal of the waste generated by the planned work.

3.2. HISTORICAL PERSPECTIVE

The LACBWR Decommissioning Plan was approved on August 7, 1991. Because the licensing history of LACBWR spans a period that includes several decommissioning regulation changes, The D-Plan has been revised to the LACBWR Decommissioning Plan and Post-Shutdown Decommissioning Activities Report (D-Plan/PSDAR).

In a letter dated October 8, 2015 (Reference 1), Dairyland and LaCrosseSolutions, LLC (Solutions) requested Nuclear Regulatory Commission (NRC) consent to transfer Dairyland's possession, maintenance and decommissioning authorities, under Possession Only License No. DPR-45, from Dairyland to Solutions. The NRC approved this transfer in a letter dated May 20, 2016 (Reference 2). The revised cost estimates presented reflect those developed by Solutions.

After the balance of the site is remediated and the as-left radiological conditions are demonstrated to be below the unrestricted use criteria specified in 10 CFR 20.1402, the licensed area will be reduced to a small area around the ISFSI and Possession Only License No. DPR-45 will be transferred back to Dairyland.

3.3. PREVIOUS DAIRYLAND COST ESTIMATES

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 (Reference 3), 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.

3.0 ESTIMATE OF EXPECTED DECOMMISSIONING COSTS (cont'd)

An improved site-specific decommissioning cost study was performed by Sargent & Lundy (S&L) in 1994 and provided basis for the updated 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 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.

3.0 ESTIMATE OF EXPECTED DECOMMISSIONING COSTS (cont'd)

3.4. SOLUTIONS DECOMMISSIONING COST ESTIMATE

The decommissioning cost estimate presented herein represents the projected costs to complete the remaining decommissioning work as of October 1, 2015. This estimate was prepared by SOLUTIONS based upon an assessment of the remaining work and incorporating experience gained while performing similar decommissioning tasks including the ongoing decommissioning of the Zion Nuclear Power Station (ZNPS) through the work of its subsidiary ZionSolutions LLC.

The decommissioning cost estimate includes application of contingency, as specific provision for unforeseeable elements of cost within the defined project scope. Contingencies are particularly important where previous experience has shown that unforeseeable events, which may increase costs, are likely to occur. The contingency, as used in this estimate, does not account for price escalation and inflation in the costs of decommissioning over the remaining project duration.

The site-specific decommissioning cost estimate presents a breakdown of all costs associated with completing the decommissioning and unrestricted release of the LACBWR site, other than the area bounded by the ISFSI. The estimate includes the costs required to accomplish unrestricted release and restore the site to a safe and stable condition as well as the operation of the ISFSI until the site and the remaining ISFSI are transferred back to Dairyland.

3.5. COST ESTIMATE DESCRIPTION AND METHODOLOGY

The cost estimates include consideration of regulatory requirements, contingency for unknown or uncertain conditions, and the availability of low and high-level radioactive waste disposal sites. The methodology utilized to develop the cost estimate follows the basic approach presented in "Guidelines for Producing Commercial Nuclear Power Plant Decommissioning Cost Estimates, (Reference 4)" which uses a unit cost factor approach for estimating the decommissioning activity costs. It also includes the use of site specific information when available (e.g., hourly labor rates, and commodities).

The updated DPC estimate completed in March 2013 has been utilized to obtain site-specific commodity quantities for this estimate. The commodity weights and estimated unit cost factors were applied, which take into consideration the current decommissioning approach and schedule, to arrive at an updated cost estimated to decommission LACBWR. Dairyland and Solutions also utilized 25 years of corporate experience in planning and scheduling as well as the latest available industry experience (e.g., information from the decommissioning of ZNPS).

The estimate does not include the transfer of spent fuel, which has been previously transferred to an ISFSI facility, the security costs for the ISFSI facility, or the removal of certain large components and decommissioning work previously completed.

Additionally, Dairyland and Solutions performed a contingency and risk analysis so that the potential additional costs due to expected but undefined risks and uncertainties could be addressed and included in the cost estimate.

The resulting information was then compiled into a decommissioning cost estimate. The following sections provide a summary of those results.

3.0 ESTIMATE OF EXPECTED DECOMMISSIONING COSTS (cont'd)

3.6. SUMMARY OF THE SITE-SPECIFIC DECOMMISSIONING COST ESTIMATE

The overall remaining decommissioning cost (including scope risk contingency) is estimated to be $[redacted] Million (in current year dollars), with a base estimated cost of $[redacted] Million, plus a scope risk contingency of $[redacted] Million. The cost estimates include provisions for cost escalation based upon the following assumptions:

- All estimated costs including labor, staff, materials, equipment, professional services, waste transportation and disposal are in 2015 dollars.
- Although all costs are in current year dollars, the project baseline does include provisions to escalate costs based on the Consumer Price Index for all Urban Customers – U.S. City Average All Items, Not Seasonally Adjusted (CPI-U NSA).
- The associated Class A radioactive waste management costs are covered by existing fixed-price contracts with EnergySolutions. Therefore, the waste management costs for these items are well known and not likely to vary due to waste volume uncertainties.
- No costs for Class B/C waste are included in the estimate, as all materials classified as B/C waste were previously removed by Dairyland.

The cost estimate includes the costs for radiological decommissioning and site restoration. A summary of the cost for each part of the decommissioning program is provided in Table 3-1.

Table 3-1 Cost Summary for Radiological Decommissioning and Site Restoration

	Radiological Decommissioning	Site Restoration[1]	Total Project
Performance Baseline	$[redacted] Million	$2.6 Million	$[redacted] Million
Contingency	$[redacted] Million	$0.3 Million	$[redacted] Million
Total	$[redacted] Million	$2.9 Million	$[redacted] Million

Note 1: Site restoration is included for completeness, but not required for license termination funding purposes.

3.7. ISFSI MANAGEMENT

All spent nuclear fuel elements from LACBWR have been transferred from the FESW to dry cask storage in the ISFSI. Solutions will assume responsibility for the ISFSI Site, including security requirements. Solutions has entered into a "Company Services Agreement" with Dairyland, pursuant to which Dairyland will provide operations, maintenance, access control, and security services to and for the ISFSI site. Dairyland is responsible for the costs relating to the ISFSI and those costs are not included in this decommissioning estimate.

3.8. SITE RESTORATION COSTS

Solutions acknowledges that the costs to restore the LACBWR property are not considered by the NRC staff as part of decommissioning costs. Nevertheless, there is significant interest by many stakeholders in these costs and they are presented herein. The estimated cost for the anticipated work scope is $2.6 Million. A contingency of $0.3 Million is estimated, bringing the total cost to $2.9 Million.

3.0 ESTIMATE OF EXPECTED DECOMMISSIONING COSTS (cont'd)

Overall, that work scope includes removal of any remaining hazardous materials, demolition of remaining structures, backfilling of any open excavations or void spaces, and final grading and stabilization against erosion. The estimated costs include the labor, equipment, materials, professional services and fees needed to conduct the work. In general, most of this work is anticipated to be performed by contractors however, the estimated cost also includes all of the program support activities and services necessary to manage and safely carry out the work.

3.9. DECOMMISSIONING FUND

Decommissioning costs will be paid for with funds from the site's Nuclear Decommissioning Trust (NDT) fund. The decommissioning of the LACBWR site ISFSI will be undertaken by Solutions and will be financed separately to the NDT account amount identified here for decommissioning of the LACBWR site.

The project cash balance of the NDT identified for the decommissioning of the LACBWR site, as agreed to by Solutions, and held in trust by the Owner trustee as of October 1, 2015 was $[redacted] Million.

Based on a time phased cash flow analysis of the radiological decommissioning and site restoration costs, and assuming NDT returns at an annual 2% real, after tax rate of return, the required minimum funding assurance amount to fund the future radiological decommissioning costs equals $[redacted] Million, which is below the $[redacted] Million available balance described above.

This NDT position, together with the $[redacted] Million Surety Bond payable to the NDT, provides for sufficient funding and financial assurance for the completion of the decommissioning of the LACBWR site.

Additionally, although not relied upon here, Solutions parent EnergySolutions has agreed with Dairyland to provide a performance guaranty defined in the LACBWR Decommissioning Agreement submitted as part of the license transfer application.

This PSDAR will not be updated for minor changes in anticipated decommissioning costs. However, the status of the decommissioning funding will continue to be reported to the NRC in accordance with 10 CFR 50.75(f)(1), "Reporting and recordkeeping for decommissioning planning." Additionally, Solutions will inform the NRC in writing of any significant schedule and decommissioning cost changes per 10 CFR 50.82(a)(7), and provide an updated site-specific estimate of remaining decommissioning costs with the license termination plan per 10 CFR 50.82(a)(9)(ii)(F).

3.10. REFERENCES

1. Letter from Dairyland Power Cooperative to the Nuclear Regulatory Commission, for Order Approving License Transfer and Conforming Administrative License Amendments, dated October 8, 2015.
2. Marlayna Vaaler, U.S. Nuclear Regulatory Commission, Letter to Barbara Nick, Dairyland Power Cooperative, "Order Approving Transfer of License for the La Crosse Boiling Water Reactor from the Dairyland Power Cooperative to LaCrosseSolutions, LLC and Conforming Administrative License Amendment" dated May 20, 2016.
3. Letter from the Nuclear Regulatory Commission to Dairyand Power Cooperative, Order to Authorize Decommissioning and Amendment No. 66 to Possession Only License No. DPR-45 for La Crosse Boiling Water Reactor, dated August 7, 1991.

3.0 ESTIMATE OF EXPECTED DECOMMISSIONING COSTS (cont'd)

4. T.S. LaGuardia et al., Guidelines for Producing Commercial Nuclear Power Plant Decommissioning Cost Estimates, AIF/NESP-036, May 1986.