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DAIRYLAND POWER COOPERATIVE

LA CROSSE BOILING WATER REACTOR

DECOMMISSIONING PLAN

AND

POST-SHUTDOWN DECOMMISSIONING ACTIVITIES REPORT

REVISED

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4.0 PLANT POST-FUEL ACCIDENT ANALYSIS

4.1 OVERVIEW

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

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

- Airborne pathway; and
- Liquid discharge pathway.

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

4.2 RADIONUCLIDE RELEASE LIMITS APPLIED IN ANALYSIS

4.2.1 Limits Applied to Postulated Airborne Release

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

1. The limits of 10 CFR 100.11 that specify that the total radiation dose to an individual at the exclusion area boundary for two hours immediately following onset of a postulated fission product release shall not exceed 25 rem (whole body) and 300 rem (thyroid; see Section 4.3.2.6).
2. The EPA protective action guidelines (PAGs – Reference 4.8.2) that specify the potential offsite dose levels at which actions should be taken to protect the health and safety of the public. The EPA PAG limits include a total effective dose equivalent (TEDE) of 1 rem.

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

4.2.2 Limits Applied to Postulated Liquid Releases

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

4.3 POST-FUEL ACCIDENT ANALYSIS ASSUMPTIONS

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

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

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

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

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

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

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

4.3.1.1 Portion of Total Radioactivity Assumed Releasable Via the Airborne Pathway

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

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

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

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

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

4.3.1.2 Portion of Total Radioactivity Assumed Releasable Via the Liquid Pathway

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

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

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

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

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

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

4.3.2 Additional Assumptions – Postulated Airborne Release

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

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

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

4.3.2.2 Terrain Height Above Grade

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

4.3.2.3 σ_Y and σ_Z at Distances Less Than 100 Meters

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

4.3.2.4 Pasquill Stability Class

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

4.3.2.5 Rem vs Rad

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

4.3.2.6 Thyroid Dose

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

1. There is no radioiodine present in the LACBWR site (non-ISFSI) radionuclide inventory; and

2. The CEDE dose conversion factor (DCF) for the only other thyroid significant nuclide, Co-60, is approximately 3.5 times greater than the thyroid DCF. Since the CEDE DCF is

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

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

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

1. 0 to1800 seconds – Radioactivity builds up over the first 30 minutes when the fumigation X/Q is used. During this period the inlet concentration is determined by the fumigation X/Q.
2. 1800 to 36,000 seconds – Radioactivity is exhausted over the remaining 9-½ hours when the non-fumigation X/Q is used. During this period the inlet concentration is 0.0 because the elevated release X/Q is 0.0.

4.3.2.8 Radionuclide Data

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

4.3.2.9 Atmospheric Release Inputs

The following values were used in the analysis.

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

4.3.3 <u>Additional Assumptions – Postulated Liquid Release</u>

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

4.3.3.1 Retention Tank Release, Dilution, and Mixing

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

4.3.3.2 Duration of Retention Tank Rupture Release to Thief Slough

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

4.3.3.3 Thief Slough and G-3 Outfall Flow

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

4.3.3.4 Liquid Release Inputs

The following values were used in the analysis.

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

4.4 SUMMARY OF ANALYSIS RESULTS

4.4.1 Postulated Airborne Release

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

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

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

4.4.2 Postulated Liquid Release

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

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

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

4.5 RADIOLOGICAL OCCUPATIONAL SAFETY

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

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

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

4.6 OFFSITE RADIOLOGICAL EVENTS

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

4.7 NON-RADIOLOGICAL EVENTS

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

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

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

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

4.8 REFERENCES

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

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

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

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

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

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

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

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

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

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

Notes:

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

Table 4-2 Summary Results of Liquid Release Analysis

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

Notes:

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

5.0 ENVIRONMENTAL IMPACT

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

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

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

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

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

The environmental impact of all completed or planned LACBWR decommissioning and dismantlement activities is SMALL as determined by the GEIS. LACBWR decommissioning is specifically evaluated in the GEIS. As stated in the GEIS, licensees can rely on information in this Supplement as a basis for meeting the requirements in 10 CFR 50.82(a)(6)(ii). Site-specific potential environmental impacts not determined in the GEIS are:

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

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