SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

RELATED TO AMENDMENT NO. 302

TO FACILITY OPERATING LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT 1

DOCKET NO. 50-285

Date
January 31, 2024

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ADAMS	Agencywide Documents Access and Management System
ALARA	As Low as (is) Reasonably Achievable
AMCG	Average Member of the Critical Group
AMSL	Above Mean Sea Level
ANL	Argonne National Laboratory
ARERR	Annual Radiological Effluent Release Report
AREOR	Annual Radiological Environmental Operating Report
BcDCGL	Base Case Derived Concentration Guideline Limit
BcSOF	Base Case Sum of Fractions
BFM	Basement Fill Model
bgs	Below-ground surface
Bq	Becquerel
Bq/L	Becquerel per liter
C-14	Carbon-14
CFR	*Code of Federal Regulations*
cfs	Cubic feet per second
Ci	Curie
cm	Centimeter
cm^2	Square centimeter
cm^3	Cubic centimeter
Co-60	Cobalt-60
cpm	Counts per minute
Cs-137	Cesium-137
CSM	Conceptual Site Model
D&D	Dismantlement and Decontamination
DA	Deconstruction Area
DCGL	Derived Concentration Guideline Limit
$DCGL_{EMC}$	DCGL that represents the same dose to an individual for residual radioactivity in a smaller area within a survey unit, by taking into account the area factor
$DCGL_W$	DCGL for the average residual radioactivity in a survey unit
DFS	Decommissioning Funding Status
dpm	Disintegrations per minute
dpm/100 cm^2	Disintegrations per minute per 100 square centimeters
DQA	Data Quality Assessment
DQO	Data Quality Objective
DRP	Discrete Radioactive Particle
DSAR	Defueled Safety Analysis Report
DSR	Dose to Source Ratio
EMC	Elevated Measurement Comparison
EPA	U.S. Environmental Protection Agency
ER	Environmental Report
ETD	Easy to detect
Eu-152	Europium-152
FCS	Fort Calhoun Station, Unit 1

FOV	Field of View
FR	*Federal Register*
FSAR	Final Safety Analysis Report
FSS	Final Status Survey
ft	Feet
ft^2	Square foot
ft^3/yr	Cubic feet per year
g/cm^3	Grams per cubic centimeter
H-3	Tritium
HPGe	High Purity Germanium
HSA	Historical Site Assessment
HTD	Hard to Detect
ISFSI	Independent Spent Fuel Storage Installation
ISGRS	In-Situ Gamma Ray Spectrometry
ISOCS	In-Situ Object Counting System
K_d	Distribution coefficient
km	Kilometer
km/hr	Kilometers per hour
L/kg	Liters per kilogram
LAR	License Amendment Request
LTP	License Termination Plan
m	Meter
m/s	Meters per second
m^2	Square meters
m^3	Cubic meters
m^3/s	Cubic meters per second
m^3/yr	Cubic meters per year
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual (NUREG-1575)
MDA	Minimum Detectable Activity
MDC	Minimum Detectable Concentration
mph	Miles per hour
mrem	Millirem
mrem/yr	Millirem per year
mSv	MilliSievert
mSv/yr	MilliSievert per year
MW	Monitoring Well
Ni-63	Nickel-63
NIST	National Institute of Standards and Technology
NORM	Naturally Occurring Radioactive Material
NRC	U.S. Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
OpDCGL	Operational Derived Concentration Guideline Limit
$OpDCGL_B$	Operational DCGL for basement structural surfaces

OPPD	Omaha Public Power District
OpSOF	The sum of fractions for each survey unit calculated using the OpDCGL
ORISE	Oak Ridge Institute for Science and Education
pCi/g	PicoCuries per gram
pCi/L	PicoCuries per Liter
pCi/m^2	PicoCurie per meters squared
QA	Quality Assurance
QAPP	Quality Assurance Project Plan
QC	Quality Control
RA	Radiological Assessment
RAI	Request for Additional Information
RASS	Remedial Action Support (In-Process) Surveys
REMP	Radioactive Effluent Monitoring Program
RESRAD	The RESRAD family of computer codes is a regulatory tool for evaluating radioactively contaminated sites, specifically designed to help determine the allowable RESidual RADioactivity in site cleanup
RESRAD-ONSITE	A specific code within the RESRAD family of computer codes designed to estimate radiation doses and risks from RESidual RADioactive materials in soils.
RG	Regulatory Guide
ROC	Radionuclide of Concern
ROCs	Radionuclides of Concern
SER	Safety Evaluation Report
SIRWT	Safety Injection Refueling Water Tank
SOF	Sum-of-Fractions
Sr-90	Strontium-90
SRP	Standard Review Plan (NUREG-1700, Revision 2)
Sv	Sievert
TEDE	Total Effective Dose Equivalent
TSD	Technical Support Document
USCS	Unified Soil Classification System
yr	Year

1 INTRODUCTION. BACKGROUND, AND REGULATORY EVALUATION

1.1 Introduction

By letter dated August 3, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21271A178), as supplemented by letters dated January 13, 2022 (ML22034A602), June 15, 2022 (ML22167A199), February 27, 2023 (ML23060A197), August 24, 2023 (ML23236A478), and December 6, 2023 (ML23346A153), Omaha Public Power District (OPPD) submitted a license amendment request (LAR) to the U.S. Nuclear Regulatory Commission (NRC) for Fort Calhoun Station, Unit 1 (FCS) to approve FCS's License Termination Plan (LTP) and add License Condition 3.D to include LTP requirements and establish criteria for determining when changes to the LTP require prior NRC approval.

The LTP submittal dated August 3, 2021, was Revision 0 of the LTP, was submitted as a supplement to the FCS defueled safety analysis report (DSAR) and was accompanied by a requested license amendment that, if approved, in addition to approving the LTP, would add License Condition 3.D, which would establish the criteria for when changes to the LTP require prior NRC approval. License Condition 3.D would also include LTP requirements. OPPD submitted the LAR in accordance with the provisions of 10 CFR Part 50.90.

The NRC accepted the proposed LAR and LTP for review by letter dated February 10, 2022 (ML22038A675), and published a notice of consideration of the proposed LAR and a finding of no significant hazards consideration determination in the *Federal Register* on March 22, 2022 (87 *FR* 16249). The notice offered a 30-day comment period and a 60-day period to request a hearing or to petition for leave to intervene. No comments, hearing requests or petitions for leave to intervene were received.

In accordance with 10 *Code of Federal Regulations* (CFR) 50.82(a)(9)(iii), the NRC is required to hold a public meeting near the relevant site after the licensee submits an LTP so that NRC staff can discuss the NRC's review of the LTP. This meeting was announced with ads in the Washington County [Nebraska] Enterprise and Pilot-Tribune newspapers on July 8, 2022, and July 12, 2022, respectively, and in the *Federal Register* on July 8, 2022 (87 *FR* 40870). The meeting was held on July 13, 2022, at the Blair Public Library & Technology Center in Blair, Nebraska. The *Federal Register* Notice also included a 60-day opportunity for the public to provide comments on the LTP in accordance with 10 CFR 50.82(a)(9)(iii). An NRC summary of that public meeting was issued on August 15, 2022 (ML22224A043) and includes a summary of questions from the public and the NRC staff's answers. No other public comments were received on the LTP.

By letters dated December 30, 2022 (ML22357A065) and June 5, 2023 (ML23256A279) NRC sent requests for additional information (RAIs) to OPPD requesting that OPPD provide additional information for the NRC's review of the LTP license amendment request. OPPD responded to those RAIs by letters dated February 27, 2023 (ML23060A197), and August 24, 2023 (ML23236A478), respectively. To consolidate the changes to the LTP represented in the RAI responses, and make other changes and clarifications to the LTP, OPPD submitted Revision 1 of the LTP by letter dated December 6, 2023 (ML23346A152). Subsequent references to the LTP in this safety evaluation report (SER) will refer to Revision 1 of the LTP.

1.2 Background

FCS was a one-unit Combustion Engineering, pressurized light water moderated and cooled nuclear power reactor facility in Fort Calhoun, Washington County, Nebraska. While operating, the FCS site consisted of approximately 540 acres on the west bank of the Missouri River, approximately 19.4 miles north of Omaha, Nebraska. OPPD also had a perpetual easement on approximately 475 acres of land on the east bank of the river directly opposite the plant buildings. The site is bounded on the north and south by farmland and on the west by U.S. Highway 75.

The NRC issued an operating license on August 8, 1973. The plant officially went online on September 1, 1973, with commercial operation beginning September 26, 1973. Other milestones related to the operational history of FCS, the significance of which are discussed later in this safety evaluation, or are significant to the decommissioning timeline, are as follows:

- In 1993, the site experienced a flooding event as a result of historically high Missouri River water levels.
- In the spring of 2011, Missouri River flooding of the site caused an extended outage of the reactor.
- In December 2013, the reactor returned to operation after recovery from the flood of 2011.
- On June 24, 2016 (ML16176A213), and updated on August 25, 2016 (ML16242A127), OPPD submitted the Certifications of Permanent Cessation of Power Operations in accordance with 10 CFR Part 50.82(a)(i).
- On October 24, 2016, the reactor was shut down for the final time.
- On November 13, 2016 (ML16319A254), the final reactor fuel off-load was completed.
- In March of 2019, the site experienced another Missouri River flood event.
- On April 10, 2019 (ML19074A301), the NRC approved the unrestricted release of approximately 120 acres of the owner-controlled area of the FCS site and 475 acres of land controlled by easements that formed part of the exclusion area of the FCS site while operating. This partial site release is described in more detail in section 9).
- OPPD began actively decommissioning FCS on April 29, 2019, and completed the transfer of all spent nuclear fuel to the Independent Spent Fuel Storage Installation (ISFSI) in May 2020.
- The LAR and LTP were submitted on August 3, 2021, and accepted for review by the NRC on February 10, 2022 (ML22038A675).
- OPPD expects to complete all activities necessary to terminate the license and release the FCS site for unrestricted use no later than the end of 2026, except for the ISFSI and

a small area surrounding the ISFSI containing the spent nuclear fuel from FCS, until its final disposition.

1.3 Regulatory Evaluation

In accordance with 10 CFR 50.82(a)(9), “[a]ll power reactor licensees must submit an application for termination of license. The application for termination of license must be accompanied or preceded by a license termination plan to be submitted for NRC approval.” The licensee has not applied for termination of the license at this time.

Under 10 CFR 50.82(a)(9)(i), the LTP must be a supplement to the final safety analysis report (FSAR), or equivalent. In accordance with 10 CFR 50.82(a)(9)(ii), the LTP must include:

(A) A site characterization;

(B) Identification of remaining dismantlement activities;

(C) Plans for site remediation;

(D) Detailed plans for the final radiation survey;

(E) A description of the end use of the site, if restricted;

(F) An updated site-specific estimate of remaining decommissioning costs;

(G) A supplement to the environmental report, pursuant to § 51.53, “Postconstruction environmental reports,” describing any new information or significant environmental change associated with the licensee's proposed termination activities; and

(H) Identification of parts, if any, of the facility or site that were released for use before approval of the LTP.

The LTP states:

> The objective of decommissioning FCS is to reduce the level of residual radioactivity to levels that permit the release of the site for unrestricted use and allow for the termination of the 10 CFR Part 50 license, excluding the ISFSI area. This LTP satisfies the requirement of 10 CFR 50.82(a)(9) to submit an LTP for U.S. Nuclear Regulatory Commission (NRC) approval. This LTP was written following the guidance in Regulatory Guide 1.179, “Standard Format and Contents for License Termination Plans for Nuclear Power Reactors” [1], NUREG-1700, “Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans” [2], NUREG-1575, “Multi-Agency Radiation Survey and Site Investigation Manual” (MARSSIM) [3], and NUREG-1757, Volume 2, “Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria” [4]. To satisfy the requirements of 10 CFR 50.82(a)(10), this LTP is accompanied by a proposed license amendment that establishes the criteria for when changes to the LTP require NRC approval.

The LTP describes OPPD’s decommissioning objective as “to conduct remediation and survey operations such that OPPD can submit a request to the NRC for the release of the site (other than the remaining licensed ISFSI facility) after meeting the unrestricted release requirements of 10 CFR 20.1402, 'Radiological Criteria for Unrestricted Use.'"

10 CFR 20.1402 states:

> A site will be considered acceptable for unrestricted use if the residual radioactivity that is distinguishable from background radiation results in a [total effective dose equivalent] TEDE to an average member of the critical group that does not exceed 25 [millirem] mrem (0.25 [milliSieverts] mSv) per year [yr], including that from groundwater sources of drinking water, and that the residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA). Determination of the levels which are ALARA must take into account consideration of any detriments, such as deaths from transportation accidents, expected to potentially result from decontamination and waste disposal.

The NRC evaluated the LTP to verify that the licensee satisfied the requirement that: 1) the LTP be submitted at least 2 years before termination of the license (50.82(a)(9)(i)); 2) the LTP includes the required parts of an LTP (50.82(a)(9)(ii)); and 3) the plans described in the LTP provide reasonable assurance that the licensee will be able to perform adequate surveys to, if performed consistent with the LTP, demonstrate compliance with the radiological criteria for unrestricted use, as specified in 10 CFR 20.1402.

As part of providing reasonable assurance that the licensee will be able to perform adequate surveys to demonstrate compliance with the unrestricted release criteria in 10 CFR 20.1402, the licensee must also provide reasonable assurance that it will be able to meet the requirements in 10 CFR 20.1501(a) and (b). Among other things, these regulations require that the licensee has performed or will perform necessary and reasonable surveys to evaluate the potential radiological hazards of the radiation levels and residual radioactivity detected.

Consistent with the above-quoted statements about which guidance the licensee used for its LTP, in conducting its review, the NRC used the guidance in:

- Regulatory Guide (RG) 1.179, Revision 2, "Standard Format and Contents for License Termination Plans for Nuclear Power Reactors" (ML19128A067);"
- NUREG-1700, Revision 2, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans" (NUREG-1700 or SRP) (ML18116A124);
- NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM), Revision 1 (ML003761445);" and,
- NUREG-1757, "Consolidated Decommissioning Guidance," – Volume 1, Revision 2; Volume 2, Revision 2; and, Volume 3, Revision 1 (ML063000243, ML22194A859, and ML12048A683 respectively).

Both RG 1.179 and NUREG-1700 identify the MARSSIM document (NUREG-1575) and NUREG-1757 as guidance for developing site characterization plans, developing remediation plans, developing site-specific derived concentration guidelines (DCGLs), demonstrating compliance with the unrestricted release criteria for license termination, and developing final survey methods and plans.

The LTP states that it was written following the guidance in RG 1.179; NUREG-1700; the MARSSIM document (NUREG-1575; and NUREG-1757, Volume 2, Revision 2. Therefore, in conducting its review, the NRC used the same guidance for determining the adequacy of the LTP for approval under 10 CFR 50.82(a)(9). Unless otherwise mentioned, these guidance documents provide an acceptable way for licensees to meet the requirements of 10 CFR 50.82(a)(9). Therefore, based on finding that an application is consistent with these guidance documents, the NRC is also able to conclude that the application satisfies the requirements of 10 CFR 50.82(a)(9).

The approval criteria for the LTP are given in 10 CFR 50.82(a)(10), which states:

> If the LTP demonstrates that the remainder of decommissioning activities will be performed in accordance with the regulations in this chapter, will not be inimical to the common defense and security or to the health and safety of the public, and will not have a significant effect on the quality of the environment and after notice to interested persons, the Commission shall approve the plan, by license amendment, subject to such conditions and limitations as it deems appropriate and necessary and authorize implementation of the LTP.

Consistent with 10 CFR 50.82(a)(10), the submitted LTP was accompanied by a proposed license amendment that would approve the LTP. Recognizing that there may be a need to make changes to the LTP following its approval by the NRC, the licensee also included, in the proposed license amendment a license condition with criteria for when changes to the LTP require NRC approval. OPPD submitted the LAR in accordance with the provisions of 10 CFR Part 50.90. In reviewing the license condition, the staff used the guidance and model license condition in appendix B of the SRP.

The proposed license condition, in the submittal dated August 3, 2021, as amended in the submittal dated February 27, 2023, would allow OPPD to make certain changes to the approved FCS LTP without prior NRC review or approval. The proposed FCS License Condition 3.D would read as follows:

3.D License Termination Plan (LTP)

The LTP NRC License Amendment No. 302 approves the
LTP, Revision 1, and the OPPD response to RAI TE 2-11 dated August 24, 2023 (ML23236A478). Changes to the LTP require prior NRC approval when the change:

(a) requires Commission approval pursuant to 10 CFR 50.59,

(b) results in significant environmental impacts not previously reviewed,

(c) detracts or negates the reasonable assurance that adequate funds will be available for decommissioning,

(d) decreases a survey unit area classification (i.e., impacted to not impacted, Class 1 to Class 2, Class 2 to Class 3, or Class 1 to Class 3 without providing NRC a minimum 14-day notification prior to implementing the change in classification),

(e) increases the dose concentration guideline limits (DCGLs) and related minimum detectable concentrations (MDCs) (for both scan and fixed measurement methods),

(f) increases in the radioactivity level, relative to the applicable DCGL, at which an investigation occurs,

(g) changes the statistical test applied to one other than the Sign test, or

(h) increases in the Type I decision error,

(i) changes the approach used to demonstrate compliance with the dose criteria (e.g., change from demonstrating compliance using DCGLs to demonstrating compliance using a dose assessment that is based on final concentration data), or

(j) changes parameter values or pathway dose conversion used to calculate the dose such that the resultant dose is lower than in the approved LTP and if a dose assessment is being used to demonstrate compliance with the dose criteria.

Based on its review of the LTP, as documented in this SER, NRC has determined that it contains the required information, as described in 10 CFR 50.82(a)(9), in adequate detail to allow for LTP approval. In addition, the NRC finds that the criteria in the proposed license condition that would be used to determine if changes to the LTP require NRC approval are equivalent to the list of LTP areas that cannot be changed without NRC approval identified in appendix B of the SRP.

As described in 10 CFR 50.82(a)(9)(i), the NRC will approve, as appropriate, the LTP is approved as a supplement to the FSAR or equivalent. For FCS, the equivalent document is the Defueled Safety Analysis Report or DSAR. The NRC's approval of the LTP, as appropriate, is predicated on the site conditions as described in the LTP and the reasonably foreseeable results of the continuing characterization of the site, the implementation of the remaining dismantlement activities and the plans for site remediation, and the LTP becoming a supplement to the DSAR. If the continuing characterization of the site, site dismantlement activities or site remediation activities find, or result in, types or quantities of residual contamination not identified in the LTP, in accordance with 10 CFR 50.59(c), the licensee must evaluate that new information against the methods of evaluating residual contamination described in the LTP to determine if a change is needed to the methods described in the LTP and if so, evaluate that change using the change criteria in the license condition approving the LTP to determine if the change needs prior approval from the NRC.

2 SITE CHARACTERIZATION

Section 2 of the FCS LTP, "Site Characterization," discusses the results of the FCS site characterization activities.

As stated in Section 2 of the LTP, "Site Characterization," the purpose of site characterization is to ensure that the final status survey will be conducted in all areas where contamination existed, remains, or has the potential to exist or remain. The applicable primary objectives of a characterization survey are discussed in section 2.4.4 of NUREG-1575, Multi-Agency Radiation

Survey and Site Investigation Manual (MARSSIM), Revision 1, dated August 2000 (ML082470583) as being:

- Determine the nature and extent of the contamination
- Collect data to support evaluation of remedial alternatives and technology
- Evaluate whether the survey plan can be optimized for using the final status survey
- Provide input to the final status survey

The SRP Acceptance Criteria for site characterization are:

- The LTP identifies all locations, both inside and outside the facility, where radiological spills, disposals, operational activities, or other radiological accidents and or incidents occurred and could have resulted in contamination. This identification should be done on a room-by-room or area-by-area basis as necessary, including equipment, laydown areas, or soils (subfloor and outside area).
- The LTP describes, in summary form, the original shutdown, and current radiological and non-radiological status of the site.
- The LTP site characterization is sufficiently detailed to allow the NRC to determine the extent and range of radiological contamination of structures, systems (including sewer systems and waste management systems), floor drains, ventilation ducts, piping and embedded piping, rubble, ground water and surface water, components, residues, and environment, including maximum and average contamination levels and ambient exposure rate measurements of all relevant areas (structures, equipment, and soils) of the site (including contamination on and beneath paved parking lots).
- The LTP identifies the survey instruments and supporting quality assurance (QA) practices used in the site characterization program.
- The LTP identifies the background levels used during scoping or characterization surveys.
- The LTP describes in detail the areas and equipment that need further remediation to allow the reviewer to estimate the radiological conditions that will be encountered during remediation of equipment, components, structures, and outdoor areas.

2.1 Summary of Characterization Survey Methods and Results

2.1.1 Characterization Survey Methodology

Section 2.1 of the FCS LTP, “Historical Site Assessment,” (HSA) and its subsections provide a summary of the HSA (ML21271A185). The HSA, completed in 2020, documents a review of the operating history of the facility, historical incidents, interviews with station personnel and operational radiological surveys. The FCS HSA was derived from a review of: records maintained to satisfy 10 CFR 50.75(g)(1), which requires the licensee to keep “[r]ecords of spills or other unusual occurrences involving the spread of contamination in and around the facility, equipment, or site”; interviews of long-tenured employees knowledgeable of site operations; records from the Nebraska Department of Environment and Energy; FCS incident files such as Condition Reports, Incident Reports, Radiological Occurrence Reports, and Licensee Event Report; FCS special survey and operational radiological survey records, engineering reports of environmental assessments and subsurface investigations at FCS; the FCS Offsite Dose Calculation Manual (ODCM); the FCS Updated Safety Analysis Reports (USAR); the FCS Spill Prevention, Control and Countermeasures Plan; the FCS Annual Radioactive Effluent Release Reports; the FCS Annual Radiological Environmental Operating Reports; the TSSD Services

Inc. 2016 “Fort Calhoun Nuclear Station Historical Site Assessment Report;” and the TSSD Services, Inc. 2016 "Fort Calhoun Nuclear Station Limited Non-Radiological Characterization Survey."

Section 2.1.3 of the LTP, “Operational History,” summarizes the history of the plant beginning in 1968 when it was granted a construction permit by the US Atomic Energy Commission, with plant construction beginning that year. The first fuel assembly was loaded into the reactor May 24, 1973. The final pre-operation fuel assembly was loaded on June 8, 1973, and the final pre-operation core verification was completed on June 9, 1973. The NRC issued an operating license on August 9, 1973. The plant officially went online on September 1, 1973, with commercial operation beginning on September 26, 1973. On June 24, 2016, as updated on August 25, 2016, OPPD submitted the Certifications of Permanent Cessation of Power Operations in accordance with 10 CFR Part 50.82(a)(1)(i). The plant went offline on October 24, 2016, and OPPD began actively decommissioning FCS on April 29, 2019. The transfer of all spent nuclear fuel to the ISFSI was completed in May 2020.

Section 2.1.4, “Incidents,” of the FCS LTP provides a summary of site incidents based on a review of plant records, with most of the events involving radiological spills and chemical spills as well as a flooding event that occurred in 2011. As part of the HSA process, OPPD divided the FCS facilities and grounds into preliminary survey areas and assigned initial area classifications based on the operational history and the incidents and processes documented for that survey unit. Survey units included Class 1, 2, and 3 structures; Class 1, 2, and 3 open land areas; and non-impacted areas. OPPD made these designations in accordance with the guidance provided in NUREG-1575, MARSSIM, Revision 1, dated August 2000 (ML082470583). To make the best use of resources for decommissioning, MARSSIM guidance suggests greater survey efforts should be made on areas that have, or had, the highest potential for contamination. Areas that have no reasonable potential for residual contamination are classified as non-impacted areas. These areas are identified through knowledge of site history or previous survey information. Class 1 areas have, or had prior to remediation, a potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiation surveys) above the DCGL for the average residual radioactivity in a survey unit ($DCGL_W$). Class 2 areas have, or had prior to remediation, a potential for radioactive contamination or known contamination, but are not expected to exceed the $DCGL_W$, based on site operating history and previous radiation surveys. Class 3 areas are not expected to contain any residual radioactivity or are expected to contain levels of residual radioactivity at a small fraction of the $DCGL_W$, based on site operating history and previous radiation surveys.

The characterization survey design is discussed in Section 2.2.3, “Survey Design,” of the LTP and its subsections. The surveys incorporated a graded approach based upon the Data Quality Objectives (DQOs) for each survey unit. Characterization surveys of Class 2 and Class 3 survey units utilize a combination of random and judgmental approaches to survey design. For Class 1 survey units, samples are primarily judgmental (as shown in Table 2-8 of the LTP “Recommended Sample Population Size”). OPPD presents scanning coverage in the survey units in Table 2-9 of the LTP, “Recommended Scan Coverage.”

OPPD performed characterization studies as part of producing its LTP. The types of characterization surveys for land areas and building surfaces are discussed in Section 2.2.4 of the LTP, “Types of Measurements and Samples.” These consisted of a combination of surface scans (beta and gamma), static beta measurements, and material samples/smears from building surfaces. For any concrete and/or asphalt-paved, open land areas that will remain after the completion of site dismantlement activities will also be subject to final status surveys (FSSs), a combination of surface scans (beta and gamma), static beta measurements, and volumetric samples. Surveys of open land areas consisted of gamma scans and the sampling of surface and subsurface soil, sediment, and surface water for isotopic analysis. Details on static measurements, beta surface scans, gamma surface scans, removable surface contamination, concrete core sampling, and material sampling were addressed in the subsections of section 2.2.4 of the LTP.

OPPD performed beta scans over accessible structural surfaces including, but not limited to, floors, walls, ceilings, roofs, asphalt, and concrete paved areas. OPPD used hand-held beta scintillation detectors (i.e., Ludlum Model 44-116) for these beta scans. OPPD performed beta surface scanning with the detector position maintained within 1.3 cm (0.5 inches) of the surface and with a scanning speed of one detector active window per second. Technicians monitored the audible response of the instrument to identify locations of elevated activity that required further evaluation.

The licensee performed static measurements to detect direct contamination levels on structural surfaces of buildings, or on concrete or asphalt paved areas; these were performed using a 126 cm^2 scintillation detector, the Ludlum Model 44-116. OPPD conducted static measurements by placing the detector on or very near the surface to be counted. OPPD acquired data over a pre-determined count time. OPPD adjusted instrument count times as appropriate to achieve an acceptable MDC for static measurements.

The licensee performed removable beta contamination or smear surveys, where applicable, to verify that loose surface contamination is less than the action level. OPPD usually took a smear for removable activity at each direct measurement location on non-asphalt type surfaces. To accomplish this, the licensee sampled a 100 cm^2 surface area using a circular cloth or paper filter, under moderate pressure. OPPD analyzed these smears for the presence of gross beta and/or gross alpha activity, as appropriate, using a proportional counting system or equivalent.

OPPD stated it performed gamma scans over open land surfaces, typically using 5 cm by 5 cm (2 inches by 2 inches) sodium iodide gamma scintillation detectors. The licensee performed gamma scans by moving the detector in a serpentine pattern, while advancing at a rate not exceeding 0.5 m (20 inches) per second. They also stated that they maintained a distance between the detector and the surface within 7.5 cm (3 inches), if possible. During the scans, technicians monitored the audible response of the instrument to identify elevated gamma flux which they flagged for further investigation.

OPPD performed volumetric sampling of concrete using a patented procedure that uses hollow drill bits to obtain exact volumes of concrete material at certain depths while utilizing a vacuum collection system. OPPD captured material from each of the incremental depths at a location in a separate container for each depth increment via use of the vacuum system.

OPPD collected surface soil (usually defined as the top 15 cm (6 inches) layer of soil) using a split spoon sampling system or by using hand trowels, bucket augers, or other suitable sampling tools. OPPD sampled subsurface soil (usually defined as below the top 15 cm (6 inches) layer in 1-meter (m) increments) by direct push sampling systems (i.e., Geoprobe) or by the use of hand augers. Sample preparation included removal of extraneous material and the homogenization and drying of the soil for analysis. Separate containers were used for each sample, and each container was custody-controlled throughout the analysis process.

Instruments used for the characterization surveys and their sensitivity are presented in Table 2-10 of the LTP, “Example of Instrument Types and Nominal MDC.” The licensee notes in Section 2.2.5.1 of the LTP, “Instrument Calibrations,” that it calibrated all data loggers, associated detectors, and other portable instrumentation used for characterization on an annual basis using National Institute of Standards and Technology (NIST) traceable sources and that the calibration of instruments is addressed in section 4.7 of the Quality Assurance Project Plan (QAPP). While OPPD primarily anticipates the on-site radiological laboratory performing gamma spectroscopy, a contracted laboratory, such as GEL laboratories, may perform radiochemical analyses. This is discussed in section 2.2.5.3 of the LTP, which also provides a summary table of the methods, MDCs, and Reporting Limits for multiple hard-to-detect (HTD) radionuclides.

2.1.2 Characterization Survey Results

Section 2.3 of the LTP, “Summary of Characterization Survey Results,” and subsections, provide a discussion of the characterization results from activities performed from 2019 through 2020. Because of the on-going decommissioning activities at the time, characterization of some areas was restricted and was to be continued during decommissioning as access allowed.

OPPD took a total of 810 concrete samples from 178 locations in various portions of the FCS site using hollow-bit drilling and analyzed 744 of these on-site by gamma spectroscopy. OPPD assumed the remaining would not have contamination because it determined outer portions of the drill sampling taken at the same location contained negligible contamination. The locations selected for the concrete sampling were biased towards areas with elevated dose rates, count rates, proximity to radiological components, or by visual observations of floor and wall surfaces that indicated potential contamination (e.g., discoloration or standing water). The breakdown of samples collected is given in section 2.3.1 of the LTP.

The licensee also conducted characterization of open land areas as described in section 2.3.2 of the LTP. OPPD analyzed each surface and subsurface soil sample with an on-site gamma spectroscopy system. In accordance with procedure, OPPD sent a minimum of 10 percent of the soil samples off-site to GEL Laboratories for full-suite ROC analysis. Sections 2.3.4.1 and 2.3.4.2 of the LTP discuss structural characterization surveys inside and outside the deconstruction area (DA). Survey results are broken down into those for Class 2 and 3 structures.

The licensee discusses surface and groundwater characterization in section 2.4 of the LTP and its subsections. The licensee noted that H-3 and Sr-90 have historically been identified sporadically in its monitoring wells in the past but that it had identified no other HTD or gamma-emitting radionuclides. Based on Annual Radiological Effluent Release Reports (ARERRs) from 2006 through 2022, concentrations of these two contaminants are sporadically but persistently detected in wells at levels slightly above detection limits when identified. In section 2.4.2. of the LTP, the licensee suggested that Sr-90 levels that a small Sr-90 release had likely occurred in the past. OPPD’s Radioactive Groundwater Protection Program (RGPP) at the plant is consistent with NEI-07-07, “Industry Groundwater Protection Initiative – Final Guidance Document” (ML19142A071), that is endorsed in NRC Regulatory Guide 4.22, “Decommissioning Planning During Operations” (ML12158A361), for sampling and analysis of monitoring wells on the site. OPPD has detected neither gamma-emitting radionuclides nor H-3 in surface water samples which OPPD sampled and analyzed consistent with the Radiological Environmental Monitoring Program (REMP).

In section 5.4.1.11 of the LTP, the licensee states that “OPPD has prepared a site procedure to describe appropriate methods and detectors suitable for identifying [discrete radioactive particles (DRPs)] and/or discrete radioactive objects that may be identified or generated during decommissioning. This procedure describes the measures for conducting contamination control surveys and the handling of DRPs if they are discovered. The procedure requirements ensure timely and prudent actions are taken in response DRPs”.

Most of the site characterization chapter of the LTP consists of figures and data tables in the pages of the attachments to this chapter. The LTP provides visual and tabulated summaries of the data obtained during the site characterization.

OPPD used the site characterization to inform the planned decommissioning activities as described in Chapter 3 of the LTP, “Identification of Remaining Site Dismantlement Activities,” which the NRC evaluates in section 3 of this SER. OPPD also used the information to plan out the FSS for the affected site areas as discussed in Chapter 5 of the LTP, “Final Status Survey Plan,” and which the NRC evaluates in section 5 of this SER.

2.1.3 Deferred Characterization Activities

2.1.3.1 Continuing Characterization

The licensee notes in Section 2.5 of the LTP, “Continuing Characterization,” that the characterization of inaccessible or not readily accessible subsurface soils, buried pipes, or concrete surfaces has been deferred. These areas will be characterized (termed “continuing characterization”) as access is gained. OPPD will determine the number and location of measurements and samples for continuing characterization using the DQOs during survey design. OPPD will capture the results of continuing characterization surveys with revisions to the Radiological Characterization Report. The following are areas at FCS where continuing characterization would occur:

- The concrete of the Spent Fuel Pool and Fuel Transfer Canal.
- The concrete walls and floor of the Under Vessel area in Containment.
- The concrete walls and floors of the 971 feet and 989 feet Above Mean Sea Level (AMSL) elevations of the Auxiliary Building.
- The concrete floors and walls of the Waste Hold-Up Tank room (Room 8).
- The concrete of the Containment Building exterior.
- The concrete of the Turbine Building basement.
- The soils underneath the concrete of the Containment Building basement, Turbine Building basement, and Spent Fuel Pool.
- The embedded floor drain systems on the 971 feet and 989 feet AMSL elevations of the Auxiliary Building.
- The embedded floor drain system on the 990 feet AMSL elevation of the Turbine Building.
- The buried pipe systems that will be abandoned in place.

The LTP states that OPPD would use continuing characterization data, along with existing data, to update the types of radionuclides present and update the variability in the radionuclide mix for both gamma-emitting and HTD radionuclides. In addition, as the decommissioning progresses, OPPD would capture data from operational events caused by equipment failures or personnel errors, which may affect the radiological status of a survey unit, in the Radiological Characterization Report. OPPD would evaluate these events and, when appropriate, store them in the characterization database. OPPD would use all characterization data in validating initial classifications and in planning for FSS.

2.1.3.2 Radiological Assessments

Section 2.5 of the LTP, "Continuing Characterization," also discusses Radiological Assessments (RAs). It states that OPPD will perform RAs in currently inaccessible soil areas that are exposed after removal of asphalt or concrete roadways and parking lots, rail lines, buried piping, or building foundation pads (slab on grade). Survey results for RAs will be presented in the relevant survey unit release records. As further noted in Section 5.2.5 of the LTP, "Radionuclides of Concern and Mixture Fractions," in order to verify that the insignificant contributor (IC) dose does not change prior to implementing the FSS, and to verify the HTD-to-surrogate radionuclide ratios used for the surrogate calculation are still valid, OPPD will obtain and analyze concrete samples and soil samples during continuing characterization (including RAs).

For continuing characterization, OPPD will analyze 10 percent of all media samples collected in a survey unit during continuing characterization for the initial suite of radionuclides from Table 5-2 of the LTP, "Initial Suite of Radionuclides," with a minimum of one sample analyzed, whichever is greater. OPPD will first analyze all samples with the on-site gamma spectroscopy system. OPPD will send the samples with the highest concentrations of gamma emitters to an accredited off-site laboratory for the analysis of the full suite of radionuclides from table 5-2 of the LTP.

2.1.3.3 Reference Areas or Materials

The site characterization did not address reference materials or areas although some measurements presented in the data tables did subtract ambient background. In Section 5.2.4 of the LTP, "Reference Areas and Materials," the licensee describes the use of the DQO process to prepare an FSS sample plan to determine whether media-specific backgrounds, ambient area background, or no background will be applied to a survey unit. OPPD will base the determination on the type of survey unit and type of measurements or samples used to demonstrate compliance. If applied, OPPD will determine media-specific backgrounds via measurements made in one or more reference areas and on various materials selected to represent the baseline radiological conditions for the site.

The LTP states that OPPD expects background at FCS to constitute a small fraction of the $DCGL_W$ based on the results of characterization surveys. For survey units with multiple materials, background data from reference areas may be subtracted from survey unit measurements (using paired observations) when the Sign test is applied as the statistical test for demonstrating compliance with the unrestricted release criteria. Alternatively, the subtraction of ambient background radiation may be considered for surface measurements. The licensee stated this will be conservative because OPPD would only subtract ambient gamma radiation, which is expected to be less than the material-specific background for the material in the area. This is because the average ambient background does not fully account for the naturally occurring radioactivity in the materials.

In response to RAI CL-5 (ML23060A197), OPPD clarified that it typically collects ambient background at various locations within the area to be surveyed. It typically establishes background as the mean of 5 one-minute static measurements from these various locations. It collects background measurements with gamma detectors at a minimum height of 15 cm (6 in) above the ground. It takes backgrounds with beta instrumentation at waist height with the detector facing away from the materials to be surveyed.

Material-specific backgrounds may be required for complex survey units or areas (e.g., floor tiles or fire extinguishers containing naturally occurring radioactive material (NORM)). In these instances, OPPD \collects the background measurements by placing the detector on its side on the material surface and recording the mean of five one-minute static measurements.

Ambient background levels can change within survey units. Examples of this occurring are when a survey is spread over several days and radon becomes more prevalent at certain points of the day or when there is a concentrated area of NORM in the survey unit. When there is a significant rise or fall in background, OPPD will establish a new ambient background using the process described above. These changes are made at the professional judgment of the LT/FSS Supervisors.

Section 6.17 of the LTP, "Above Ground Building DCGL for ROC," states that OPPD, during characterization static and scan surveys, yielded gross beta static and swab measurement results at low levels and likely at or near background. OPPD did not establish a reference area for the characterization effort because OPPD did not deem it necessary. OPPD also did not determine and net measurements.

2.1.4 NRC Evaluation of the FCS Site Characterization

The NRC reviewed the information in the FCS LTP for the FCS facility and site as well as the HSA (ML21271A609) in accordance with Section 2.2, "Site Characterization," of the SRP (NUREG-1700greg). Also, the NRC reviewed Revision 1 of the OPPD reference document TSD-21-043 entitled "Radionuclides of Concern in Support of the Fort Calhoun License Termination Plan" (ML21271A175) submitted with Revision 0 of the LTP. As described in the SRP, the purposes of the NRC's review are: (1) to ensure that the site characterization presented in the LTP is complete; and, (2) to verify that the licensee obtained the data using sufficiently sensitive instruments and proper QA procedures to obtain reliable data that are relevant to determining whether the site will meet the decommissioning limits if characterization data is used as final survey data. The acceptance criteria for section 2.2 of the SRP states that the LTP should: (1) identify all locations where activities (including spills) could have resulted in contamination; (2) summarize the status of the site; (3) be sufficiently detailed to allow a reader to determine the contamination levels; (4) identify survey instruments and practices; (5) identify background radiation levels; and, (6) describe areas and equipment that need further remediation. In addition, the staff evaluated the site characterization information using the guidance contained in NUREG-1757 and MARSSIM (NUREG-1575), which describe ways to meet the objectives of providing an adequate site characterization, as required by 10 CFR 50.82(a)(9)(ii)(A), as well as ways for the site characterization information to provide a necessary and reasonable evaluation of the site characterization and recordkeeping requirements described in 10 CFR 20.1501(a) and (b).

The LTP and HSA summarized the plant shutdown activities and past characterization efforts, and identified those known locations where spills, disposals, operational activities, or other accidents and/or incidents had occurred prior to the LTP submittal and which could have resulted in contamination within and outside of the facility. The licensee acknowledges that there were inaccessible areas that will be characterized as site decommissioning continues and that the radiological composition of contamination will be verified during such activities. The licensee evaluation of potential contaminants is summarized in TSD-21-043, "Radionuclides of

Concern in Support of the Fort Calhoun License Termination Plan," and results in 5 to 6 radionuclides of concern (ROCs) (discussed further in section 5.2 of this SER). The LTP describes the areas and equipment that need remediation in chapter 3 of the LTP and the characterization data supplied is sufficient to preliminarily determine areas approaching or exceeding the release criteria when compared to preliminary interim screening values. NRC staff reviewed the Annual "Radiological Environmental Operating Reports" (AREORs) for the years 2006 through 2022 and found that "no plant-related effects" were detected or observed in any of the off-site environmental samples, which included sediment, vegetation, surface water, groundwater, milk, air, fish, and food crops. Because no radionuclides were identified in offsite samples from the AREORs, staff finds that further radiological offsite characterization is not needed for license termination.

With regard to background, the NRC noted that the site characterization deferred the characterization of reference materials or areas. OPPD addressed his subject in the FSS section of the LTP (5.2.4, "Reference Areas and Materials") where the licensee describes considering either media-specific backgrounds, ambient area background, or no background. OPPD generally anticipates the background to be a small fraction of the $DCGL_W$ and it expects to utilize the Sign test for statistical analysis of results, meaning background subtraction is typically not needed. As a result, OPPD did not provide soil and structural reference measurements. The licensee did subtract an "ambient" background when needed from structural measurements. The licensee's discussion of ambient background being generally conservative is only true if there are no other significant sources of gamma-radiation flux in the general area (e.g., readings taken near a significantly contaminated object). Staff recognizes that this is likely the case when performing FSS surveys in and around structures after remediation although areas of elevated residual radioactivity could still be present. That said, the method discussed for determination of ambient background in structures may not be appropriate or conservative. This is because when significant radiological sources are present in the background measurement area, the ambient background is biased high resulting in lower net measurements. Therefore, during verification and confirmatory structural surveys, if a nonconservative bias in licensee's measurements is noted due to the method of considering ambient background, NRC staff anticipate applying a correction to data to negate such bias when evaluating the licensee's data during final status survey report review. NRC staff does not anticipate such a correction to create much issue with meeting the release criteria because the biases are usually small in comparison to the criteria being applied. Otherwise, the NRC found the approach taken by the licensee to determine background to be consistent with guidance in MARSSIM and NUREG-1757, Volume 2, Revision 2.

The licensee has sufficiently detailed the status of the FCS facility and site to preliminarily determine the extent and range of radiological contamination in the structures and open land areas, as well as surface water and groundwater. Staff reviewed the information in figures, tables, and text in chapter 2 of the LTP as well as the evaluations documented in TSD 21-043, "Radionuclides of Concern in Support of the Fort Calhoun License Termination Plan," and found that there were reasonable efforts made to characterize and evaluate the contamination levels in accessible portions of the site. Reasonable justification was given for areas not currently characterized and the licensee plans to perform additional characterization surveys as these areas become accessible and update its characterization report as appropriate. Therefore, the FCS LTP meets the acceptance criteria as delineated in section 2.2 of NUREG-1700. In addition, the staff evaluated the licensee's site characterization survey practices and instruments. Based on this review, the NRC determined that the licensee has met the 10 CFR 50.82(a)(9)(ii)(A) requirement that the LTP to include "[a] site characterization as well as the requirements of 10 CFR 20.1501(a) and (b).

3 REMAINING SITE DISMANTLEMENT ACTIVITIES

In accordance with the requirements of 10 CFR 50.82(a)(9)(ii)(B), the LTP must identify the remaining major dismantlement and decontamination (D&D) activities for the decommissioning of the site at the time of LTP submittal. The licensee followed the guidance of RG 1.179 and the SRP to provide that information in Chapter 3, “Identification of Remaining Site Dismantlement Activities,” of the LTP. Those activities can be undertaken before approval of the LTP pursuant to 10 CFR 50.82(a)(5) and (6), the current 10 CFR Part 50 license for FCS (License No. DPR-40), and consistent with the revised FCS Post-Shutdown Decommissioning Activities Report, dated December 16, 2019 (ML19351E355).

The guidance in RG 1.179 and the SRP describes that the LTP should include:

- A discussion of the remaining D&D tasks, decontamination techniques, and projected schedules for use in planning further decommissioning activities and for NRC to identify any inspection or technical resources needed during the remaining dismantlement activities.

- A description of the proposed control mechanisms to ensure that areas are not re-contaminated.

- Occupational exposure estimates and a characterization of the type and quantity of radioactive waste produced.
- A description of how the remaining activities are evaluated for unreviewed safety questions or against the facility’s licensing requirements.

3.1 Completed and Ongoing D&D Activities

The FCS LTP includes a discussion of the remaining D&D tasks that includes a description of: the end state of the FCS at license termination; the completed and ongoing decommissioning activities and the implementation and/or completion of tasks such as the onsite storage of the Greater Than Class C waste; the movement and storage of the remaining spent fuel from the reactor in the ISFSI; the installation of temporary enclosures and structures to handle the processing of waste produced by the D&D activities; the demolition and dismantlement of the non-radiological structures including the ones associated with the auxiliary building; and, the removal of buried utilities, tanks and piping.

3.2 Remaining Site Dismantlement Activities

The NRC reviewed the information in the FCS LTP related to the identification of remaining site dismantlement activities and evaluated those activities’ potential impacts on occupational exposure, the control mechanisms to prevent recontamination of previously remediated areas, and the estimate of the volume of radioactive waste for disposal.

3.2.1 Remaining Tasks Associated with Decontamination and Dismantlement

The LTP discusses the sequence of tasks associated with D&D, starting with the removal of non-radiological buildings within the deconstruction area (DA) and the turbine building to make room for the demolition of the radiologically contaminated buildings. Next, contaminated systems and components would be decontaminated or removed, packaged, and shipped directly to a low-level radioactive waste disposal facility. Once the radiologically contaminated systems and components are removed, radiologically contaminated buildings in the DA (e.g., Radwaste Building) would be removed. This would be followed by the reactor vessel segmentation and reactor internals removal. Buried piping would then be removed within the DA and outside the DA when available for excavation. The removal of non-radiological buildings outside the DA will then occur, along with the removal of the remaining radiological buildings within the DA (Auxiliary Building and Containment Buildings). Lastly, the temporary waste processing structures will be removed. The general project milestones for completing the remaining dismantlement activities are described in the LTP.

The LTP states that OPPD will complete a comprehensive final radiological survey to verify that residual radioactivity has been reduced to levels allowing for the release of the FCS site for unrestricted use. The LTP also said, the decommissioning activities described in the LTP would be conducted under the provisions of the OPPD Radiation Protection Program and Radioactive Waste Management Program.

3.2.2 Proposed Control Mechanisms to Ensure That Areas Are Not Re-Contaminated

The LTP states that OPPD would maintain its capability to isolate or to mitigate the consequences of radioactive release during D&D activities and that OPPD would plan work activities to minimize the spread of contamination. OPPD would contain contaminated liquids within existing or supplemental barriers and could process them by a liquid waste processing system prior to release, if necessary. The LTP identifies considerations for OPPD to evaluate for planning of decommissioning work activities to minimize the potential for spread of contamination. As described in section 3.4.2 of the LTP, OPPD will install a temporary waste loadout enclosure at the Containment Building equipment hatch and at the rail spur. These enclosures are intended to provide additional assurance that areas will not be re-contaminated. OPPD would perform routine radiological surveys within and adjacent to these enclosures, as well as along the haul road route in between the enclosures, to verify that the controls are being maintained.

3.2.3 Occupational Exposure Estimates and Radioactive Waste Characterization

The LTP stated that OPPD would develop exposure estimates and exposure controls for specific activities during detailed planning per OPPD Radiation Protection Program and written site procedures. The LTP estimated personnel exposures for various decommissioning and fuel storage activities. OPPD developed these estimates to provide site management ALARA goals and verified them through comparison with site dosimetry data. In addition, OPPD generated ALARA estimations derived from a compilation of Radiation Work Permit evaluations. From this information, OPPD estimated the total radiation exposure impact for decommissioning and spent fuel management as approximately 230 person-rem.

The LTP states that OPPD will use the OPPD Radioactive Waste Management Program to control the characterization, generation, processing, handling, shipping, and disposal of radioactive waste during decommissioning. The largest volume of low-level radioactive waste that OPPD expects to generate during decommissioning are activated and contaminated structures, systems, and components. OPPD expects the other forms of waste generated during decommissioning to 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 equipment used in the decommissioning process. OPPD estimated the total volume of radioactive waste for disposal at less than 2.5

million cubic feet and stated that most of the waste would be loaded into transport casks, gondola rail cars, or articulating bulk cars and shipped to Clive, Utah or to the licensed Waste Control Specialists facility in Andrews County, Texas. Greater Than Class C (GTCC) waste will be loaded into casks and remain on-site stored at the ISFSI.

To dispose of low-level radioactive waste (LLRW) at a licensed disposal site, the generator must characterize the physical and radiological characteristics of each container before shipment. OPPD employs a combination of analytical techniques for isotopic analysis of waste followed by waste classification (Class A, B, C, or GTCC) as defined in 10 CFR Part 61. The licensee's decommissioning LLRW classification is based on NUREG/CR-0130, "Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station," site-specific information, and industry experience. For example, the reactor vessel and internals estimated curie content at shutdown is typically derived from NUREG/CR-0130 and adjusted for radioactive decay and differing mass of the components.

3.2.4 <u>Licensee evaluation of remaining decommissioning activities against unreviewed safety questions</u>

In accordance with 10 CFR 50.82(a)(9)(i) the LTP was submitted as a supplement to the FCS DSAR. As stated in the LTP, OPPD will conduct the decommissioning activities at FCS in accordance with the DSAR, the facility operating license (License No. DPR-40, Docket No. 50-285), all associated technical specifications, and the requirements of 10 CFR 50.82(a)(6) and (a)(7). OPPD states in the LTP that the remaining activities do not involve any unreviewed safety questions or changes in the technical specifications for FCS. However, if an activity requires prior NRC approval under 10 CFR 50.59(c)(2), or a change to the technical specifications or license, a submittal will be made to the NRC for review and approval before implementing the activity in question. The licensee's process for determining when changes to the FCS licensing basis require NRC approval is subject to NRC inspection.

3.3 Remaining Site Dismantlement Conclusions

The LTP summarizes the remaining site D&D activities and techniques to be used and includes information regarding those areas and equipment that need further radiological remediation and an estimate of radiological conditions that the licensee may encounter. The licensee provided a description of the major remaining components of radiologically contaminated plant systems and specific equipment remediation considerations and a general schedule for completion of the D&D milestones. The LTP also describes the proposed control mechanisms to ensure remediated areas are not re-contaminated. The LTP also includes estimates of associated occupational radiation doses, projected volumes of radioactive waste, and a description of radioactive waste characterization. The licensee provides a description of how the remaining D&D activities are evaluated against the licensing basis for the plant and the requirements in the license for any unreviewed safety questions.

Based on this review, the NRC determined that the licensee has identified, in sufficient detail, the remaining dismantlement activities necessary to complete decommissioning of the facility to meet the requirement in 10 CFR 50.82(a)(9)(ii)(B) and provide a basis for use in planning further decommissioning activities to allow for NRC to identify any inspection or technical resources needed during the remaining dismantlement activities.

4 PLANS FOR RADIOLOGICAL SITE REMEDIATION

In accordance with 10 CFR 50.82(a)(9)(ii)(C) and the guidance of RG 1.179, Chapter 4, “Remediation Plans,” of the LTP details the remediation methods and techniques that the licensee will use to demonstrate that the facility D&D activities will be conducted in accordance with established Radiation Protection (RP), Safety, and Waste Management programs. The FCS LTP also states OPPD frequently audits these programs and procedures for technical content and compliance. The licensee stated that the remaining residual radioactivity will satisfy the As Low as [is] Reasonably Achievable (ALARA) criterion in 10 CFR 20.1402.

4.1 FCS Remediation Plans

Chapter 4 of the FCS LTP provided a discussion of the licensee’s plans for completing radiological remediation of the site. The licensee plans to remediate the site, including structures and systems that remain on the site, to the unrestricted use criteria, specified in 10 CFR Part 20, Subpart E, of 0.25 mSv/yr (25 mrem/yr) for all pathways. The licensee stated that the remaining residual radioactivity will also satisfy the ALARA criterion in 10 CFR 20.1402, the demonstration of which involves a determination of whether it is feasible to further reduce residual radioactivity to levels below those necessary to meet the dose criterion (i.e., to levels that are ALARA). The FCS LTP summarized the radiological control program that the licensee will implement for the control of radiological contamination associated with the decommissioning and remediation, as well as the radiation protection methods and control procedures employed to address the impact of dismantlement and remediation activities. Since the licensee will be remediating the site to the unrestricted release criteria of 10 CFR 20.1402, no submission regarding a restricted end use of the site is required. Therefore, the licensee has complied with the requirement of 10 CFR 50.82(a)(9)(ii)(E).

According to Section 4.2.1 of the LTP, “Structures and Piping,” six above-grade buildings will remain at the time of license termination: Training Building, Flex Building, Owner Controlled Area (OCA) Entrance Building, Switchyard 3451 Old Building, Switchyard 3451 New Building, and the 1251 Control and Switchgear Building. Except for the 354 kilovolt (kv) and 161 kv lines to the station, the Switchyard also will remain. Section 5.4.1.9 of the LTP, “Survey Considerations for Above-Ground Buildings and Miscellaneous Structures,” stated above-grade structures that will remain at the time of license termination will undergo FSS using the acceptable screening values for building surface contamination from table H.1 of Appendix H, “Criteria for Conducting Screening Dose Modeling Evaluations,” of NUREG-1757, Volume 2, Revision 2. If the number and identification of above-ground buildings to remain changes, the NRC will be notified. Minor solid structures such as, but not limited to, the Switchyard, telephone poles, fencing, culverts, duct banks and electrical conduit will be included in the FSS of the open land survey unit in which they reside.

The LTP stated that the basements of the Turbine Building, Containment Building, Auxiliary Building, and Intake Structure will remain with all interior walls removed. The exception is the Turbine Building where the pedestals will remain within the interior of the basement. Basements will be demolished to at least three feet (0.92 m) below grade corresponding to 1,001 feet AMSL. OPPD will remove all concrete inside the liner from the interior of the Containment Building, including activated and contaminated concrete and embedded piping. A below-grade electrical vault from the Radwaste Processing Building will remain following demolition. After completion of decontamination and dismantlement and the FSS, basements will be backfilled with stockpiled and characterized fill material to grade level (approximately 1,004 feet AMSL). Remediation techniques that will be used before backfilling for the structural surfaces below the

1,001 feet AMSL elevation included washing, wiping, pressure washing, vacuuming, scabbling, chipping, and sponge or abrasive blasting, and for concrete removal hydraulic-assisted, remote-operated, articulating tools.

The licensee plans to use an elevated measurement comparison (EMC), as described by MARSSIM Section 2.5.1.1, “Small Areas of Elevated Activity,” to establish a soil $DCGL_{EMC}$ action level, and will remove and dispose of soil contamination above the site-specific $DCGL_{EMC}$ as radioactive waste. Soil remediation equipment specified in the LTP included shovels, backhoe, and track hoe excavators, soil dredges and vacuum trucks. As practical, when the remediation depth approaches the soil interface region between unacceptable and acceptable contamination, a squared edge excavator bucket design or similar technique will be used to minimize the mixing of contaminated soils with acceptable lower soil layers as would occur with a toothed excavator bucket. The licensee committed to using excavation safety and environmental control procedures to remediate radiologically contaminated soils. The licensee augmented the excavation safety and environmental control procedures with other procedural requirements to ensure the licensee maintains adequate erosion, sediment, and air emission controls during soil remediation.

Section 4.3, “Remediation Activities Impact on the Radiation Protection Program,” of the LTP stated that the current RP Program is adequate to safely control the radiological aspects during decommissioning and does not present new challenges above those encountered during normal plant operations and refueling. The licensee stated it was not requesting approval of any changes to the existing RP Program as a part of the LTP. In section 4.3 of the LTP, the licensee states the program is protective of occupational personnel expected to encounter radiological hazards from decommissioning of a reactor facility, ensures the protection of the public from radiological hazards, and makes sure occupational, effluent, and environmental dose from radiological materials remain ALARA. Furthermore, it also stated that during decommissioning, contamination reduction techniques and engineering controls are used to mitigate the spread of contamination and reduce personnel exposure to radiation and contamination.

The licensee provided its ALARA analysis process and conclusions for soil and remaining basement structures in Section 4.4, “ALARA Evaluation,” of the FCS LTP. The licensee’s formulas for calculating the remediation levels followed the guidance provided in NUREG-1757, Volume 2, Revision 2, Appendix N, “ALARA Analyses,” which describes acceptable methods for determining when further reduction of residual radioactivity is required to concentrations below the levels necessary to satisfy the 0.25 mSv/yr (25 mrem/yr) dose criteria. For soil, the licensee concluded, with a simplified analysis, that the cost of disposing of excavated soil as low-level radioactive waste is greater than the benefit of removing and disposing of soil with residual radioactivity concentrations less than the dose criterion.

According to the LTP, most of the residual radioactivity remaining in structures after the activated and contaminated concrete removal from the Containment Building, will be located in the Auxiliary Building basement. Thus, the ALARA assessment for remediated basement structures focused on the Auxiliary Building floors of the 971 feet and 989 feet AMSL elevations, which bounded the ALARA assessment for other buildings. Three BFM dose scenarios (in situ, drilling spoils, and excavation) were calculated separately and summed to evaluate concrete scabbling and shaving remediation. For the basement structures that will remain, the ALARA analysis showed that further remediation of concrete beyond that to demonstrate compliance with the 0.25 mSv/yr (25 mrem/yr) dose criterion is not required.

4.2 NRC Evaluation of the Radiological Site Remediation Plan

The NRC reviewed the information in the FCS LTP for the FCS facility and site using Section 2.4, “Remediation Plans,” of NUREG-1700. As described therein, the purposes of the NRC’s review are to ensure the LTP: (1) addresses any changes in the radiological controls to be implemented to control radiological contamination associated with the remaining decommissioning and remediation activities; (2) discusses in detail how facility and site areas will be remediated to meet the proposed residual radioactivity levels (DCGLs) for license termination; and (3) includes a schedule that demonstrates how and in what time frame the licensee will complete the interrelated decommissioning activities.

The FCS LTP discussed in detail how the licensee intends to remediate the FCS facility and site to meet the proposed residual radioactivity levels (DCGLs) for license termination, including a summary of the removal and remediation tasks planned for surface and subsurface soil and concrete structures at the site, as well as the techniques associated with these tasks. The LTP also included a summary of the radiation protection methods and control procedures that will be employed during the remaining decommissioning activities. Table 3-2, “General Project Milestones” of the LTP contained a schedule that demonstrates how the licensee intends to complete the interrelated decommissioning activities.

The LTP provided the details of the licensee’s ALARA analyses to ensure compliance with the criterion specified in 10 CFR 20.1402. The NRC evaluated the licensee’s ALARA analyses for soils and remaining basement structures and concluded that they were acceptable for determining situations when the costs for additional dose reduction below the regulatory release criterion exceed the calculated benefit value, and therefore comply with the ALARA criteria of 10 CFR 20.1402. The NRC verified the licensee’s calculations in the ALARA analysis presented in Table 4-5 of the LTP, “ALARA Analysis for the Auxiliary Building Basement,” where all derivations were significantly greater than unity demonstrating that the criteria being applied were considered ALARA. During the staff review of the ALARA calculations provided in section 4.4 of the LTP, using equations from appendix N of NUREG-1757, Volume 2, Revision 2, staff identified an error in the $Cost_{PDose}$, which is included in the total cost ($Cost_T$). The total cost is then used to determine the $Cost_{ALARA}/DCGL_W$, which, if greater than 1, validates that further remediation beyond that required to demonstrate compliance with the 0.25 mSv/yr (25 mrem/yr) dose criterion is not justified. The error in $Cost_{PDose}$ resulted in a slightly lower $Cost_{ALARA}/DCGL_W$; however, the NRC calculated $Cost_{ALARA}/DCGL_W$ produced the same outcome (i.e., $Cost_{ALARA}/DCGL_W>1$).

Therefore, the FCS LTP meets the acceptance criteria as delineated in SRP section 2.4, which is one way to meet the regulatory requirements. Based on this review, the NRC finds the licensee has provided a sufficiently detailed discussion of its radiological site remediation plans for the remaining decommissioning activities, as required by 10 CFR 50.82(a)(9)(ii)(C) (requiring the LTP to include “[p]lans for site remediation”).

5 FINAL RADIATION SURVEY PLAN

In accordance with the requirements of 10 CFR 50.82 (a)(9)(ii)(D) and the guidance of RG 1.179, the LTP, Chapter 5, “Final Status Survey Plan,” provides a description of the methods to be used in planning, designing, conducting, and evaluating the FSS at FCS. The FSS plan describes the final survey processes that will be used to demonstrate that the FCS site complies with the radiological criteria for unrestricted use specified in 10 CFR 20.1402. Additional regulations applicable to FSS are also found in Subpart F, “Surveys and Monitoring,” of 10 CFR Part 20 at 10 CFR 20.1501(a) and (b).

The final status survey is the radiation survey performed after an area has been fully characterized and remediated and the licensee believes that the area is ready to be released. The purpose of the final status survey is to demonstrate that the site, or portion thereof, under consideration meets the radiological criteria for license termination in Subpart E of 10 CFR Part 20.

According to the guidance contained in RG 1.179, which OPPD states the LTP follows, a licensee should include the following items, which are not meant to be all inclusive, in the final radiation survey plan:

a. Describe the methods proposed for surveying all equipment, systems, structures, and soils, as well as a method for ensuring that sufficient data are included for a meaningful statistical survey.

b. Describe the methods the licensee will use to establish background radiation levels. Include a discussion of variances in background radiation that can be expected (e.g., between structures constructed of different materials).

c. Describe the QA program to support both field survey work and laboratory analysis. Address the QA organization; training and qualification requirements; survey instructions and procedures, including water, air, and soil sampling procedures; document control; control of purchased items; inspections; control of survey equipment; handling, storage, and response checks; shipping of survey equipment and laboratory samples; disposition of nonconformance items; corrective action; QA records; and survey audits, including methods to be used for reviewing, analyzing, and auditing data.

d. Describe the verification surveys and evaluations used to support the delineation of radiologically affected (contaminated) areas and unaffected (uncontaminated) areas.

e. Identify the major radiological contaminants.

f. Discuss methods used for addressing hard-to-detect radionuclides.

g. Describe access control procedures to avoid recontamination of clean areas.

h. Identify survey units having the same area classification.

i. Describe scanning performed to locate small areas of elevated concentrations of residual radioactivity.

j. Discuss levels established for investigating significantly elevated concentrations of residual radioactivity.

k. Describe the reference coordinate system established for the site areas.

The NRC staff compared the information in chapter 5 of the FCS LTP against the acceptance criteria in Section 2.5, “Final Radiation Survey Plan,” of NUREG-1700. As described therein, the purpose of the NRC staff’s review is to ensure the LTP includes (1) the “Information To Be

Submitted," as described in Section 4.4, "Final Status Survey Design," of NUREG-1757, Volume 2, Revision 2; (2) the following information: identification of the major radiological contaminants; methods used for addressing HTD radionuclides; access control procedures to control recontamination of clean areas; description of the QA program, and; methods for surveying embedded and buried piping; and (3) a final survey plan that meets the evaluation criteria defined in section 4.4.1.2 of NUREG-1757, Volume 2, Revision 2.

Chapter 5 of the FCS LTP notes that the FSS plans were based on multiple guidance documents including: RG 1.179; NUREG-1700, Revision 2; NUREG-1757, Volume 2, Revision 2; NUREG-1575 (MARSSIM), Revision 1; NUREG-1505, "A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys," Revision 1, (ML061870462); NUREG-1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions," Revision 1, (ML003676046), and; NUREG-1576 (MARLAP).

The scope of the FCS FSS plan includes the RA of all impacted backfilled structures, excavations created as a result of the removal of basement structures, buried piping, open land areas, and above-grade buildings that will remain following decommissioning. The licensee indicates in the LTP that it is their intention to release for unrestricted use the impacted open land areas, select backfilled basement structures, buried and embedded piping and remaining above grade buildings from the 10 CFR Part 50 license, with the exception of the immediate area surrounding the ISFSI. The FCS FSS plan does not address non-impacted areas that were identified through site history or previous survey information and were previously released from the license.

5.1 Final Status Survey Design

Section 5, "Final Status Survey Plan," of the LTP, and its subsections, presents the FCS FSS design. Section 5.1.3 of the LTP, "Scope," notes that the FSS Plan includes the RA of impacted structures, systems and land areas that will remain following decommissioning. Impacted areas are defined as areas with a reasonable possibility of containing residual radioactivity in excess of natural background or fallout levels. OPPD intends to release for unrestricted use the impacted open land areas, remaining below-ground basements, above-ground buildings, and buried piping from the 10 CFR 50 license. Figure 3-1 of the LTP provides the location of the remaining basements, above-ground buildings, and buried piping that will remain at the time of license termination. The adjacent areas that were classified as non-impacted and previously released from the 10 CFR Part 50 license, will not be subject to FSS. The ISFSI, including the ISFSI Operations Facility, which will remain a licensed area, will also not be subject to FSS. In addition, Section 5.4.1.10 of the LTP, "Groundwater," addresses the plan for the assessment of groundwater.

Section 5.1.4 of the LTP, "Summary of the FSS Process," notes that the primary objectives of the FSS are: verify the survey unit classification; demonstrate that the potential dose from residual radioactivity in each survey unit is below the release criterion; and demonstrate that the potential dose from small areas of elevated activity, when combined with other residual radioactivity in a survey unit, is below the release criterion. The FSS process consists of four principal elements: planning, design, implementation, and assessment.

Survey planning is discussed in Sections 5.1.4.1 and 5.2 of the LTP, "Summary of Survey Planning," and "Final Status Survey Planning," respectively. The survey planning includes review of the HSA and other pertinent characterization to establish survey unit classification and the ROCs. OPPD developed DCGLs and area factors for the various impacted media as discussed in section 6 of this SER. In areas where OPPD knows remediation is required (prior to the performance of FSS) or identified during FSS, it will perform a remedial action support survey (RASS)to verify that remediation was successful, and that the area is suitable for initiating or completing FSS. In areas where remediation is not required, OPPD will perform RAs to determine if the radiological conditions in an area or structure are suitable for performing FSS, update the existing characterization survey results with additional survey data (continuing characterization), and assess the radiological conditions of excavations performed to expose/remove or install buried components Prior to implementation of FSS, OPPD will establish isolation and control methods in survey units to ensure that radioactive material is not reintroduced into the area from ongoing decommissioning activities and to maintain the “as-left” radiological and physical conditions of the area. An inspection of a survey unit will be performed to ensure the area is suitable for turnover to perform FSS.

Section 5.3 of the LTP, “Final Status Survey Design,” and its subsections outline the general approach taken for the final status survey design. The licensee developed an FSS design using the DQO process as outlined in appendix D of MARSSIM. The seven steps for the DQO process are discussed in Section 5.2.1, “Data Quality Objectives,” and its subsections of the LTP. The general approach in MARSSIM for FSS involves a minimum number of measurements or samples being taken within a survey unit so that the nonparametric statistical tests used for data assessment can be applied with adequate confidence. For the FCS FSS, OPPD chose the Sign test as the nonparametric statistical test because OPPD expects background to constitute a small fraction of the $DCGL_W$ based on results of characterization surveys. As discussed above, OPPD will not subtract background when demonstrating compliance with the exception of subtraction of ambient background for buildings and piping as discussed in Section 5.2.4 of the LTP, “Reference Areas and Materials.” The decision errors that OPPD will use as part of the DQO process are discussed in Section 5.3.1.1 of the LTP, “Decision Errors.” OPPD plans to utilize a Type 1 error rate of 0.05 and a Type 2 error rate of 0.05 although it notes that it may modify the Type 2 error rate. The lower bound of the grey region (LBGR) will be set at the mean concentration of residual radioactivity, if known. If no other information is available, OPPD may initially set the LBGR to half of the operational DCGL (OpDCGL). In that instance, OPPD may set the LBGR as low as the MDC for the specific analytical technique. OPPD will estimate the standard deviation of samples within the survey unit from RA/RASS surveys, characterization surveys, or investigation data when available. If not available, the survey design will utilize a coefficient of variation of 30% as a reasonable value for the standard deviation consistent with MARSSIM guidance. The optimal value for relative shift is noted as being between 1 and 3.

The classification of survey unit areas is meant to establish the level of survey effort needed for FSS. The initial survey units with classifications are listed in tables 2-1 through 2-5 of the LTP and shown in figures 2-1 and 2-2 of the LTP. The area limits for survey units are specified in table 5-1 of the LTP, “FCS Survey Unit Surface Area Limits,” which OPPD states is consistent with MARSSIM. For FCS basements, OPPD has developed a “Basement Fill” exposure pathway model as discussed in section 6 of this SER. The licensee states that it may exceed the survey unit sizes in the table specified for basements, because it uses the BFM, while maintaining sample/measurement frequency (i.e., sample/unit area) consistent with the maximum area for the classification of the survey unit as listed in the table. The percent coverage for scan surveys in survey units is specified in Table 5-20 of the LTP, “Recommended Survey Coverage for Open Land Areas and Structures,” which NRC staff note is consistent with guidance in MARSSIM, section 5.5.3.

OPPD will establish a reference grid to facilitate scanning and setting measurement/sampling locations. At a minimum, OPPD will define a benchmark for each survey that will serve as an origin for documenting survey efforts and results. OPPD will collect systematic measurements or samples in Class 1 or 2 survey units in a pattern or grid as discussed in Section 5.3.6.2 of the LTP, “Systematic Sampling and Measurement Locations.” OPPD will establish a random starting point using a random number generator with a grid having either a triangular or square pattern. For Class 3 survey units, OPPD will randomly select each sampling or measurement location. Trained and qualified personnel will perform survey measurements and collect samples. This will include surface scanning, static measurements, gamma spectroscopy of volumetric materials, and in-situ gamma spectroscopy.

5.2 Radionuclides of Concern

5.2.1 Radionuclides of Concern During Decommissioning

Section 5.2.5 of the FCS LTP, “Radionuclides of Concern and Mixture Fractions,” discusses the licensee’s anticipated ROCs and fractional makeup to be encountered during decommissioning. OPPD provided the initial list of potential radionuclides in tables 2-7 and 5-2 of the FCS LTP “Initial Suite of Radionuclides.” OPPD initially considered the ROCs in FC-18-002, “Potential Radionuclides of Concern During the Decommissioning of Fort Calhoun Station,” Revision 2, (ML22034A595), which describes the basis for an initial suite of potential ROCs for the decommissioning of the FCS facility and site. Guidance documents that OPPD referenced for this assessment included NUREG/CR-3474 and NUREG/CR-4289. In TSD 21-043, “Radionuclides of Concern in Support of the Fort Calhoun License Termination Plan,” OPPD further refined the radionuclides of concern and presents mixture percentages. OPPD considered the analytical results from the sampling of various media at the site (e.g., 57 concrete samples and 47 soil samples obtained during site characterization). Based on the elimination of some of the potential neutron activation products, noble gases, and radionuclides with a half-life of less than 2 years, OPPD refined its initial suite of potential ROCs for the decommissioning of FCS. OPPD evaluated the relative doses and insignificant contributor dose and selected the ROCs for Auxiliary Building embedded pipe in TSD 22-110 “Evaluation of Auxiliary Building Embedded Pipe ROCs and Surrogate DCGLs at Fort Calhoun” (Attachment 5 of ML23346A152).

The suite of dose significant ROCs at FCS for use during decommissioning is presented in Table 5-3, “Dose Significant Radionuclides and Renormalized Mixture Fractions,” of the FCS LTP. OPPD established three groupings of radionuclide mixtures, one for the Containment Building, another for the remainder of the site with the exception of the embedded pipe in the Auxiliary Building, and one for the embedded pipe in the Auxiliary building. The licensee noted that the containment building had a relatively large amount of C-14, which is why that ROC mixture was separately considered (i.e., approximately 83% of the total activity was C-14 inside the containment building vs only 5% for the Auxiliary Building, Turbine Building and Radioactive Waste Processing Building [AB/TB/RWPB] per table 6-3 of the LTP). The licensee also noted that this would need close monitoring to ensure no cross contamination occurred during decommissioning. The licensee additionally states that “there is no indication from the characterization data or operational history that contamination is present in FCS soils” and “that the AB/TB/RWPB radionuclide mixture is assumed to apply to soil, buried pipe and basement fill.” The LTP states that any uncertainties in the AB/TB/RWPB mixture applied to the soil and as well with the buried piping would likely not cause a significant variation in the dose in relation to the 0.25 mSv/yr (25 mrem/yr) dose criterion.

The licensee further delineated and established a listing of ROCs and their applicable release criteria for various media and portions of the site in tables 5-5 through 5-15 of the LTP. These tables provide ROCs for specific media at the site:

- ROCs for the CB walls and floors are C-14, Co-60, Sr-90, Cs-137, and Eu-152.
- ROCs for the CB mix fraction soil, fill material and buried pipe scenarios are C-14, Co-60, Cs-137, and Eu-152.
- ROCs for Other walls/floors are C-14, Co-60, Sr-90, Cs-137, and Eu-152
- ROCs for AB embedded pipes are C-14, Co-60, Ni-63, Sr-90, and Cs-137
- ROCs for TB embedded pipes are C-14, Co-60, Sr-90, Cs-137, and Eu-152
- ROCs for AB/TB/RWPB mix fraction soil, fill material, and buried pipe scenarios are C-14, Co-60, Cs-137, and Eu-152.
- ROCs for groundwater are C-14, Co-60, Cs-137, Eu-152, and Sr-90.

5.2.2 Insignificant Contributors

The process OPPD utilized to establish the mixture fractions and initial suite of radionuclides is discussed in TSD 21-043, “Radionuclides of Concern in Support of the Fort Calhoun License Termination Plan,” TSD 22-110 “Evaluation of Auxiliary Building Embedded Pipe ROCs and Surrogate DCGLs at Fort Calhoun,” as well as Section 6.15 of the LTP, “Insignificant Contributor Radionuclide Dose and Selection of ROC.” The licensee developed mixture fractions using the 75th percentile of isotope fractions as related to Cs-137 (i.e., the ratio of any particular radionuclide concentration to that of Cs-137 concentration in the same sample). When normalized to a 0.25 mSv/yr (25 mrem/yr) level, the licensee could then establish the potential dose contribution from each initial ROC. Table 6-3 of the LTP, “Initial Suite Radionuclide Mixture Using the ‘75th Percentile of the Cs-137 Fractions’ Approach,” lists the initial suite of radionuclides along with their mixture fractions. The licensee determined several insignificant dose contributors based upon the guidance contained in Section 3.3, “Insignificant Radionuclides and Exposure Pathways,” of NUREG-1757, Volume 2, Revision 2.

OPPD calculated the relative dose fraction for each initial suite radionuclide for each DCGL (tables 6-7, 6-13, 6-15, 6-16, 6-17, 6-18, 6-21, and 6-28 of the LTP) and mixture (Containment, AB/TB/RWPB, and AB embedded pipe). OPPD also evaluated the IC radionuclide dose from the 0.25 mSv/yr (25 mrem/yr) concentrations for the less likely but plausible scenarios. For each DCGL/mixture pair, the evaluation uses the radionuclide mixture fractions of the initial suite of radionuclides to select the radionuclides that are ICs and determine the aggregate dose from the eliminated radionuclides. The radionuclides remaining after the IC radionuclides are eliminated are the ROCs that will undergo detailed assessment during FSS.

OPPD accounted for the IC dose fractions by adjusting the DCGLs for each ROC. For basement walls/floors and turbine building embedded pipe, the ROCs were found to be Cs-137, Co-60, Eu-152, Sr-90, and C-14. The aggregate dose percentage from the IC radionuclides in basement floors/walls ranges from 0.52% (IC dose fraction of 5.2E-03) to 1.25% (IC dose fraction of 1.25E-02). The IC dose fraction for turbine building embedded pipe is 4.94% (IC dose fraction of 4.94E-02) and the IC dose fraction for the auxiliary building embedded pipe is 5.04% (IC dose fraction of 5.04E-2). The ROCs for soil and buried pipe are Cs-137, Co-60, Eu-152, and C-14. The aggregate dose percentage from the IC radionuclides in soil ranges from 0.82% (IC dose fraction of 8.2E-03) to 2.39% (IC dose fraction of 2.39E-02). The maximum value of 2.39% applies to the soil DCGL for a 1 m source thickness. The IC dose for buried pipe is 1.05% (IC dose fraction of 1.05E-02). The ROCs for soil are also assigned to aboveground buildings. The ROCs for fill material are also Cs-137, Co-60, Eu-152, and C-14 (see section 6.19 for discussion of fill DCGLs) with an IC radionuclide dose fraction of 8.88% (IC dose fraction of 8.88E-02).

Also, as stated in Section 5.2.5 of the LTP, “Radionuclides of Concern and Mixture Fractions,” the licensee has also committed to characterizing previously inaccessible areas as part of the continuing characterization process as decommissioning progresses. To verify that the IC dose does not change prior to implementing the FSS, and to verify the HTD to surrogate radionuclide ratios used for the surrogate calculation are still valid, the licensee will obtain and analyze concrete samples and soil samples during continuing characterization, which includes RAs. In response to RAI TE-5 (ML23060A197), OPPD states that it will analyze at least 10% of all media samples collected in a survey unit during continuing characterization and RA/RASSs for the initial suite of radionuclides from table 5-2 of the LTP to verify insignificant contributor dose and HTD to surrogate ratios are unchanged from those reported in TSD-21-043, “Radionuclides of Concern in Support of the Fort Calhoun License Termination Plan.” OPPD will include the results of the IC dose and surrogate ratio evaluations in the release record of the subject survey unit.

5.2.3 NRC Evaluation of Radionuclides of Concern and Insignificant Contributors

The NRC staff evaluated the licensee’s proposed ROCs, insignificant contributors, and use of surrogates in accordance with the regulatory guidance and acceptance criteria contained in NUREG-1757, Volume 2, Revision 2, Appendix I, “Technical Basis for Site-Specific Dose Modeling Evaluations,” and section 2.5 of NUREG-1700. NRC staff found that the licensee’s use of decay corrected activity and the 75th percentile values to determine mixture fractions and insignificant contributors as well as using 95th percentile values to establish surrogate ratios to be reasonable in the LTP. This is consistent with guidance in section 3.3 of NUREG-1757, Volume 2, Revision 2, and MARSSIM Section 4.3.2, “DCGLs and the Use of Surrogate Measurements.”

The licensee’s intent to continue verification of ROCs and insignificant contributors is reasonable because parts of the site that were inaccessible during site characterization will be characterized upon the licensee gaining access to those areas during the deconstruction process and because cross contamination may occur as the containment structure is demolished and removed. The continuing characterization of previously inaccessible areas and the RAs and RASSs supporting decommissioning may identify non-conservative variations in the radionuclide mixtures or the identification of significant ROCs other than what was previously determined. For example, NRC staff noted that OPPD excluded Sr-90 as a ROC from the soil mixtures and related materials for both the Containment Building (CB) and the Auxiliary Building, Turbine Building and Radioactive Waste Processing Building (AB/TB/RWPB). Upon review of TSD 21-043, “Radionuclides of Concern in Support of the Fort Calhoun License Termination Plan,” staff found that OPPD incorporated Sr-90, which does not occur naturally and has a half-life of 29 years, into the insignificant contributor portion for these media because it, along with the other insignificant radionuclides, contributes less than 10% of the overall dose from residual radioactivity consistent with NRC guidance. Regardless, Rev 0 of the LTP stated that groundwater monitoring suggests that a Sr-90 release may have occurred in the past based on the sporadic, statistical identification of Sr-90 in groundwater. Given this potential indication of past Sr-90 release, the NRC requested (RAI TE2-2) more support for why Sr-90 is not considered an ROC in the deconstruction area. In response to RAI TE2-2 (ML23236A478), the licensee maintained that Sr-90 was appropriately considered an insignificant contributor and that the LTP mistakenly suggested that a Sr-90 release may have occurred at the site. While

NRC staff acknowledge the licensee's statements, staff also considered that the statistical identification of Sr-90 contamination in groundwater could still indicate a release of Sr-90 may have occurred at the site, most likely below the primary building areas. For this reason, in section 5.2.5 of Revision 1 of the LTP, the licensee stated that it will analyze all continuing characterization samples at locations within the footprints of the Containment Building, Auxiliary Building, Radwaste Building, and Turbine Building, for Sr-90 in addition to gamma spectroscopy analysis.

The NRC staff also noted that the licensee's plan to continue characterization in inaccessible areas and perform RA and RASS surveys to support decommissioning activities will result in verification that the ROCs and insignificant contributors are appropriate throughout the decommissioning process. Based on the discussion provided in this section of the SER, as well as additional details provided in response to RAI TE-5 (ML23060A197) wherein the licensee discusses its insignificant contributor and surrogate continuing evaluation process, the NRC staff finds that the ROCs and insignificant contributors identified in the LTP are reasonable and appropriate for the FCS site and consistent with guidance found in MARSSIM and NUREG-1757 and demonstrate compliance, in part, with 10 CFR 50.82 (a)(9)(ii)(D).

5.3 Release Criteria for the FCS Facility and Site

5.3.1 LTP Discussion and Commitments for FCS Release Criteria

Section 5.2.6, "Release Criteria," and its subsections of the FCS LTP discusses the FSS criteria OPPD will use to demonstrate compliance with the radiological criteria for unrestricted use, as specified in 10 CFR 20.1402. First, OPPD established and adjusted the Base Case DCGLs (BcDCGLs) that it uses to demonstrate compliance with the 0.25 mSv/yr (25 mrem/yr) unrestricted release criterion to allow consideration of insignificant contributor (IC) dose. IC dose is discussed in Section 6.15 of the LTP, "Insignificant Contributor Radionuclide Dose and Selection of ROC," and ranges from 0.52% to 8.88% of the 0.25 mSv/yr (25 mrem/yr) release criteria depending on the medium considered. The OpDCGLs are a fraction of the Base Case DCGLs based on the dose fraction for each affected medium as presented in Table 5-4 of the LTP, "*a priori* Dose Fractions for End State Media," and will be used to guide remediation efforts and to ensure that the summation of dose from each medium is 0.25 mSv/yr (25 mrem/yr) or less after all FSSs are completed. The OpDCGL will be used as the DCGL for the FSS design of the survey unit (calculation of surrogate DCGLs, investigation levels, etc.). Details of the OpDCGLs derived for each dose component and the basis for the applied *a priori* dose fractions are provided in chapter 6 of the LTP which details the approach, modeling parameters, and assumptions used to develop the BcDCGLs. The assessment of DCGL development is discussed in Section 6 of this SER, "Compliance with Radiological Criteria for License Termination."

The tables in section 5.2.6 of the LTP and its subsections provide BcDCGLs as well as OpDCGLs, most of which have been adjusted for IC dose. OPPD developed these DCGLs for various media it anticipated to encounter, including:

- floor/walls of the Containment Building,
- floor/walls of the Auxiliary Building/Turbine Building/Intake Structure/Circulating Water Tunnels,
- surface soil,
- subsurface soil,
- buried piping,

- embedded pipe for various floors of the auxiliary building and turbine floor, and
- aboveground buildings.

Table 5-15 of the LTP, "Dose Conversion Factors for Existing Groundwater," provides dose conversion factors (mrem/yr per pCi/L) for residual radioactivity in groundwater. The licensee states that it will include only positively detected ROC groundwater monitoring results in the dose calculation for groundwater. OPPD notes that the assumed dose fraction of 2% that it allocated to groundwater may be exceeded so long as the total potential dose from all media under consideration does not exceed 0.25 mSv/yr (25 mrem/yr).

The licensee has defined soils both as "surface soil" extending to 0.15 m depth and also as a layer of soil beginning at the surface but extending to a depth of 1 m. It did this to allow for flexibility in demonstrating compliance if it encounters contamination deeper than 0.15 meters. OPPD determined DCGLs for soil for both situations. Based on characterization data and historical information, the licensee does not expect to encounter a source term geometry that is comprised of a clean surface layer of soil over a contaminated subsurface soil layer.

Section 5.2.6.7 of the LTP, "Dose from Fill Material," discusses dose from soil material used to backfill excavations and basement structures being left on-site. This section provides BcDCGLs, adjusted for insignificant contributor dose, for this material. The LTP notes that dose from fill soil above the 1001 feet AMSL elevation (3 feet below grade) is the dose attributable to the Class 3 open land soil and will be calculated using surface soil DCGLs. OPPD will calculate the dose from fill below 1001 feet AMSL using the DCGLs given in Table 5-19 of the LTP, "Base Case DCGLs for in situ Fill ($BcDCGL_F$) Adjusted for IC Dose," (reproduced from table 6-37 of the LTP). OPPD assigned no OpDCGLs to the in-situ fill materials.

The LTP also states that the BcDCGLs, which OPPD established for the average residual radioactivity in a survey unit, are equivalent to a $DCGL_W$ (as typically described in MARSSIM guidance). As such, the $DCGL_W$ can be multiplied by an area factor to derive the $DCGL_{EMC}$, which represents a radionuclide concentration in a smaller area that should similarly equate to a potential dose of 0.25 mSv/yr (25 mrem/yr). OPPD will only apply the $DCGL_{EMC}$ to Class 1 open land soil survey units as those are the only survey units in which OPPD anticipates elevations in residual radioactivity above the $DCGL_W$. The usage of area factors for soil is described in Section 5.2.6.5, "Soil Area Factors," and Section 6.20, "Soil Area Factors for ROC," of the FCS LTP. Area Factors for soil are presented in Tables 5-17 and 5-18 of the LTP, "Soil Area Factors for ROC 0.15 m Thickness," and "Soil Area Factors for ROC 1.0 m Thickness," respectively. The licensee doesn't provide area factors for structures because, as it states in section 6.11.10 of the LTP, "the surface area of excavated concrete, after spreading on the ground surface, is assumed to be large such that area factors do not apply."

OPPD is applying the default screening values from table H.1 in appendix H of NUREG-1757, Volume 2, Revision 2, to above-grade structures that will remain on the FCS site at the time of license termination. If applicable, OPPD will base any screening values not addressed in table H.1 on values derived or taken from Table 5.19 of NUREG/CR-5512, "Residual Radioactive Contamination from Decommissioning," Volume 3, "Parameter Analysis," dated October 1999 (ML082460902). The concentrations listed in disintegrations per minute (dpm) per 100 square centimeters (dpm/100 cm^2) in table 5.19 are equivalent to a dose of 0.25 mSv/yr (25 mrem/yr). OPPD has developed BcDCGLs as well as OpDCGLs for basement structures that OPPD will backfill (the Containment Building, Auxiliary Building, Turbine Building, Intake Structure, and Circulating Water Tunnels) as discussed in sections 5.6.1.1 and 5.6.1.2 of the LTP, respectively.

Since multiple ROCs exist at the FCS site, OPPD will use a sum-of-fractions (SOF) (unity rule) for FSSs to ensure that the total dose from all ROCs does not exceed the criterion for unrestricted release. The licensee notes in Section 5.5.3.1, "Sum of Fractions," of the FCS LTP that the use and application of the unity rule will be in accordance with MARSSIM Section 4.3.3, "Use of DCGLs for Sites with Multiple Radionuclides."

OPPD will ultimately demonstrate compliance, as stated in section 5.2.6 of the FCS LTP, "through the summation of dose from seven distinct media in the end-state (basement floor/walls [includes steel liner in Containment and penetrations], basement embedded pipe, basement fill, above-ground buildings, soils, buried pipe, and existing groundwater)." Final demonstration of the compliance dose determination is discussed in section 6.1 of this SER.

The requirements in 10 CFR 20.1402 also require that the residual radioactivity levels be ALARA. The licensee evaluates the various remediation activities planned in chapter 4 of the LTP to determine whether the planned activities are ALARA. The licensee evaluated soil remediation and concrete remediation. The licensee noted that "the vast majority of residual radioactivity remaining in the structures after the concrete is removed from the Containment Building basement, will be located in the Auxiliary Building basement. Therefore, the ALARA assessment for the remediation of basement structures will focus on the floors of 971 feet and 989 feet AMSL elevations of the Auxiliary Building, as this is the location where the greatest benefit of concrete remediation could be achieved. An ALARA assessment of the Auxiliary Building basement floor will bound ALARA assessments for the other buildings which would use the same methods (and cost estimate) but remove less contamination." In both the soil and concrete removal ALARA evaluations, OPPD showed the ratio of the residual radioactivity concentration that is ALARA to that of the DCGLs to be significantly greater than unity demonstrating that meeting the release criteria (DCGLs) would be considered ALARA as discussed in section 4.1 of this SER.

5.3.2 Use of Surrogates

The use of surrogate radionuclides is discussed in section 5.2.6.2 of the LTP. Surrogates are used to reduce costs for measurement and analysis when an established relationship between HTD and ETD radionuclides is present. A surrogate DCGL allows the DCGLs specific to HTD radionuclides in a mixture to be expressed in terms of a single radionuclide that is more readily measured or ETD. The ETD or measured radionuclide is called the surrogate radionuclide. OPPD will apply the surrogate approach to basement walls/floors, embedded pipe, buried pipe, and aboveground buildings at FCS. OPPD established surrogate ratios for HTD ROCs (C-14, Ni-63 and Sr-90) based on their ratios to Cs-137 as shown in table 5-16 of the LTP. The licensee evaluated the containment building separately from the AB/TB/RWPB areas of the site due to relatively high concentrations of C-14 in the containment building. OPPD similarly evaluated the AB embedded pipe separately. The ratios were derived in TSD-21-043, "Radionuclides of Concern in Support of the Fort Calhoun License Termination Plan," and TSD-22-110, "Evaluation of Auxiliary Building Embedded Pipe ROCs and Surrogate DCGLs at Fort Calhoun," and are the 95th percentile of the ratios found in the samples taken from concrete. For the Auxiliary Building embedded piping, the average ratio was selected for the final ratios due to a lower sample size used in the analyses. OPPD will modify the Cs-137 DCGL, in accordance with section 4.3.2 of MARSSIM, if the sampling analysis/measurement excludes the HTD ROCs (C-14 and Sr-90) to ensure these contaminants are adequately accounted for. The licensee states that actual analytical data may be utilized if available and adequate for that purpose. The licensee also notes that C-14 will be directly measured in soil samples as opposed to inferred by surrogate measurement as C-14 is the only HTD radionuclide it anticipates.

Equation 5-2 in the LTP (replicated below) shows the calculational method that OPPD will use to determine the surrogate DCGLs and is consistent with MARSSIM equation 4-1 utilized for the same purpose.

$$DCGL_{Surrogate} = \frac{1}{\frac{1}{DCGL_{ETD}} + \frac{R_1}{DCGL_1} + \frac{R_2}{DCGL_2}}$$

where:

$DCGL_{Surrogate}$ =	modified DCGL (or Basement Dose Factor) for surrogate ratio
$DCGL_{ETD}$ =	DCGL for easy-to-detect radionuclide
$DCGL_{HTD}$ =	(with or without 1 or 2 added) DCGL for the hard-to-detect radionuclide(s)
R_N =	Surrogate Ratio for the HTD to the ETD radionuclide(s)

As stated in Section 5.2.5 of the LTP, “Radionuclides of Concern and Mixture Fractions,” the continuing characterization process for previously inaccessible areas of the site will verify the HTD surrogate radionuclide ratios used for the surrogate calculation are still valid. OPPD will analyze ten percent of all media samples collected in a survey unit during continuing characterization for the initial suite of radionuclides from table 5-2 of the LTP, with a minimum of one sample analyzed, whichever is greater. If the analysis indicates positive results (greater than MDC) for both an HTD ROC (C-14 for concrete or soil, or Sr-90 for concrete) and the corresponding surrogate radionuclide (Cs-137), then OPPD will calculate the HTD-to-surrogate ratio. If the calculated HTD-to-surrogate ratio is less than the applicable HTD-to-surrogate ratio from table 5-16 of the LTP, then no further action is required. If the HTD-to-surrogate ratio exceeds the applicable ratio from table 5-16 of the LTP, then OPPD will take a minimum of five additional investigation samples around the original sample location. OPPD will analyze each investigation sample by the on-site gamma spectroscopy system and then send it to an off-site laboratory for HTD analysis. As with the original sample, OPPD will calculate the HTD-to-surrogate ratio for each investigation sample. OPPD will use the actual maximum HTD-to-surrogate ratio observed in any individual sample to infer HTD radionuclide concentrations in the survey units shown to be impacted by the investigation. OPPD will document the survey unit-specific HTD-to-surrogate ratio and the survey data serving as the basis for the ratio in the release record for the survey unit(s).

5.3.3 Adjusted Gross DCGLs

Section 5.2.6.3 of the LTP discusses how, for the FSS of aboveground buildings, OPPD will use adjusted gross DCGLs ($DCGL_{AG}$). This is done because radionuclide-specific data is not typically acquired during static measurements; instead, field instruments will typically measure gross alpha or gross beta/gamma emissions from the surface during static measurements. The $DCGL_{AG}$ will be calculated using Equation 5-3 of the LTP (replicated below) which is consistent with Equation 4-4 in MARSSIM for the same purpose.

$$DCGL_{AG} = \frac{1}{\left[\left(\frac{f_1}{DCGL_1}\right) + \left(\frac{f_2}{DCGL_2}\right) + \dots\left(\frac{f_i}{DCGL_i}\right)\right]}$$

where:

$DCGL_{AG}$	=	Adjusted Gross DCGL in units of dpm/100 cm^2
$DCGL_i$	=	DCGL for detectable radionuclide in units of dpm/100 cm^2
f_i	=	Mixture fraction of detectable radionuclides

5.3.4 NRC Evaluation of Release Criteria, Use of Surrogates, and Adjusted Gross DCGLs

The NRC staff evaluated the licensee's proposed release criteria, use of surrogates, and adjusted gross DCGLs using the regulatory guidance and acceptance criteria contained in NUREG-1757, Volume 2, Revision 2, Section 4.1.4, "Release Criteria," and section 2.6 of the SRP. The licensee provides DCGLs for each medium (except GW where Dose Conversion Factors are provided), provides reasonable surrogate ratios for HTD ROCs and discusses how it will develop surrogate measurement criteria, and also presents area factors that will be utilized if residual radioactivity in soil at concentrations exceeding the OpDCGLs is encountered. The methods discussed for applying the release criteria, surrogate DCGLs, and adjusted gross DCGLs, are consistent with guidance in MARSSIM and NUREG-1757.

As previously evaluated in section 4 of this SER, the DCGLs were demonstrated to be ALARA which is consistent with statements in appendix N of NUREG-1757, Volume 2, Revision 2, that "in light of the conservatism in the building surface and surface soil generic screening levels…licensees who remediate building surfaces or soil to the generic screening levels…do not need to provide reanalyses to demonstrate that these screening levels are ALARA," and that "an ALARA analysis is not needed for soil removal to meet unrestricted release at or below a dose criterion of 0.25 mSv (25 mrem) per year."

Based on the discussion provided in this section of the SER, as well as additional details provided in chapters 4 (regarding how the remediation will be ALARA), and 6 (regarding demonstration of meeting the criteria), the NRC staff finds that the methods discussed for evaluating residual radioactivity and comparison to the dose criteria are reasonable and demonstrate, in part, compliance with 10 CFR 50.82(a)(9)(ii)(D).

5.4 Decommissioning Support Surveys

Decommissioning support surveys include: the HSA, characterization surveys, continued characterization surveys, RA surveys, RASSs, etc. The HSA, Characterization Surveys, continued characterization surveys, and RA surveys—which were discussed in Section 2 of this SER, "Site Characterization." The LTP notes that OPPD deferred certain actions of its site characterization including for areas that were not generally accessible (continued characterization), as well as RA surveys that it will perform when land coverings are removed. In addition, another deferred characterization is that which may be required if reference materials or areas need to be assessed. These deferred characterization activities are discussed and evaluated in section 2 of this SER. As such, this section will primarily focus on RASSs and Final Status Surveys.

5.4.1 Remedial Action Support Surveys and Radiological Assessments

Section 5.1.4.1, "Summary of Survey Planning*,"* of the FCS LTP discusses decommissioning support surveys, including RA surveys and RASSs. According to the LTP, OPPD will perform RASSs in areas where remediation is required to provide real-time data collection to verify that remediation was successful, and that the area is suitable for initiating or completing FSS. OPPD will perform RA surveys, considered part of continuing characterization and further discussed in section 2.5 of this SER, to determine if the radiological conditions in an area or structure are suitable for performing FSS, update the existing characterization survey results with additional survey data (continuing characterization), and assess the radiological conditions of excavations performed to expose/remove or install buried components. OPPD will document the results of RASSs and RAs in the applicable FSS release records and, as described in Section 5.3.1.6 of the LTP, “Relative Shift,” may utilize the result to estimate the standard deviation of residual radioactivity concentrations and sampling requirements for FSS in a survey unit.

RASSs will primarily rely on scanning and sampling (or direct measurement in structures) of identified areas of interest. In basement structures, OPPD may take in situ gamma spectroscopy measurements in lieu of sampling. OPPD anticipates the methods that it will utilize during RASSs will be similar to those discussed for final status surveys in multiple sections of the LTP, including: 5.4.1.1, “Scanning;” 5.4.1.2, “Fixed Measurements;” 5.4.1.3, “Volumetric Sampling;” 5.4.1.9, “Survey Considerations for Above-Ground Buildings and Miscellaneous Structures;” and, 5.4.2.1, “Instrument Selection.” Also, as discussed in response to RAI TE-5 (ML23060A197), OPPD will analyze 10% of the media samples collected during continuing characterization and RA/RASSs for the full initial suite of radionuclides to verify IC dose and HTD to surrogate radionuclide ratios are unchanged from those reported in TSD 21-043, “Radionuclides of Concern in Support of the Fort Calhoun License Termination Plan, and TSD 22-110, “Evaluation of Auxiliary Building Embedded Pipe ROCs and Surrogate DCGLs at Fort Calhoun.”

5.4.2 NRC Evaluation of Decommissioning Support Surveys

The NRC staff evaluated the licensee’s decommissioning support survey methods in accordance with the regulatory guidance and acceptance criteria contained in NUREG-1757, Volume 2, Revision 2, Section 4.3, “Remedial Action Support Surveys,” and applicable portions of Sections 2.2 and 2.5 of the SRP, “Site Characterization,” and “Final Radiation Survey Plan,” respectively. Based on the discussion provided in this section of the SER, the NRC staff finds that the plans for decommissioning support surveys are consistent with the NRC’s guidance on RASSs, and that the plans are adequate to assist the licensee in determining when remedial actions have been successful and the FSS may commence. As such, the NRC staff finds that the licensee’s decommissioning support survey methods should be adequate to comply, in part, with 10 CFR 50.82(a)(9)(ii)(A) and (B) and provide information supporting the FSS design.

5.5 Final Status Survey Planning and Design

Survey planning is discussed in Sections 5.1.4.1 and 5.2 of the LTP, “Summary of Survey Planning,” and “Final Status Survey Planning,” respectively, as well as their subsections. The licensee considered the DQO process from MARSSIM appendix D, which includes the following actions: state the problem, identify the decision, identify inputs to the decision, define the study boundaries, develop a decision rule, specify limits on decision errors, and optimize the design for obtaining data. Section 5.2.2 of the LTP, “Classification of Survey Units,” describes the criteria for defining survey units as Class 1, 2, or 3. Section 5.2.4 of the LTP discusses Reference Areas and Materials while Sections 5.2.5 and 5.2.6, “Radionuclides of Concern and Mixture Fractions,” and “Release Criteria,” as well as their subsections, provide details for how OPPD will assess the potential dose from residual radioactivity for the various media it expects to find at the time of license termination.

Sections 5.1.4.2 and 5.3, “Summary of the Survey Design,” and “Final Status Survey Design,” and their subsections of the LTP presents the FCS FSS design. These sections note that the general approach in MARSSIM for FSS involves at least a minimum number of measurements or samples being taken within a survey unit, so that the nonparametric statistical tests used for data assessment can be applied with adequate confidence. OPPD states that the Sign test is the most appropriate test for FSS at FCS because OPPD expects background to constitute a small fraction of the $DCGL_W$ based on the results of characterization surveys. The above-mentioned sections of the LTP discuss the criteria for survey unit classification, scan coverage, number and location of surface measurements and volumetric samples. Investigation levels are discussed in Section 5.5.5.1 of the LTP, "Investigation Levels." The LTP also states that OPPD will prepare a survey map for each survey unit that depicts the locations of scan grids, as well as random, systematic or judgmental/investigative measurement and sample locations. The appropriate instruments and detectors, instrument operating modes, and survey methods used to collect and analyze data are specified in a subsection of Section 5.6 of the LTP, "Quality Assurance," which describes scanning and analytical instruments, methods, calibration, operational checks, coverage, and sensitivity for each medium and radionuclide as well as how samples will be collected, controlled, and handled.

5.5.1 <u>Statistical Considerations for FSS</u>

In Section 5.3.1 of the FCS LTP, "Sample Size Determination," the licensee indicated it will follow guidance in MARSSIM and Appendix A of NUREG-1757, Volume 2, Revision 2, "Implementing the MARSSIM Approach for Conducting Final Radiological Surveys," to determine the number of sampling and measurement locations (sample size) necessary to ensure an adequate set of data that is sufficient for statistical analysis such that there is reasonable assurance that the survey unit will pass the requirements for release. The number of sampling and measurement locations is dependent upon the anticipated statistical variation of the final dataset such as the standard deviation, the decision errors, and a function of the gray region, as well as the statistical tests to be applied. Decision errors are addressed in Section 5.3.1.1 of the LTP, "Decision Errors," and the licensee commits to setting the Type I error (which would result in the release of a survey unit containing residual radioactivity above the release criterion, or false negative) and Type II (which would result in a failure to release a survey unit when the residual radioactivity is below the release criterion, or false positive) decision errors as follows:

- the α value (probability of making a Type I error) will always be set at 0.05 (5 percent) unless prior NRC approval is granted for using a less restrictive value; and

- the β value (probability of making a Type II error) will also be initially set at 0.05 (5 percent), but may be modified, as necessary, after weighing the resulting change in the number of required sampling and measurement locations against the risk of unnecessarily investigating and/or remediating survey units that are truly below the established release criterion.

The licensee notes in Section 5.3.1.3, "Gray Region," and Section 5.3.1.6, "Relative Shift," of the FCS LTP that the gray region and relative shift calculations will utilize OpDCGLs in planning the number of required FSS measurements and the standard deviation of the data set will be initially calculated from characterization survey, RA, RASS, and/or investigation data. Section 5.3.2, "Statistical Test," of the FCS LTP discusses statistical tests to evaluate FSS results, and the licensee indicates that "the Sign Test will be implemented using the unity rule, surrogate methodologies, or combinations thereof as described in MARSSIM and Chapters 11, "Multiple Radionuclides," and Chapter 12, "Multiple Surfaces," of NUREG-1505." The licensee also states that "the Sign Test will be applied when demonstrating compliance with the unrestricted release criteria without subtracting background (other than ambient background when taking gross static measurements of buildings and piping as discussed in section 5.2.4 of the LTP).” OPPD will determine the number of sampling and measurement locations (N) that it will collect from the survey unit by establishing the acceptable decision errors, calculating the relative shift, and using table 5-5 of MARSSIM. OPPD may alternatively use MARSSIM Equation 5-2 to calculate the number of sampling and measurement locations and will round the calculated result up by 20 percent.

5.5.2 Areas of Elevated Activity and Scan Coverage

Small areas of elevated activity are discussed in Section 5.3.3, “Small Areas of Elevated Activity,” of the FCS LTP. Small areas of elevated residual radioactivity (greater than the DCGLw) are compared to the $DCGL_{EMC}$ (a DCGLw modified by an area factor to account for small areas of elevated residual radioactivity). The licensee notes that “at FCS, the consideration of small areas of elevated radioactivity will only be applied to Class 1 open land (soil) survey units as Class 2 and Class 3 survey units should not have contamination in excess of the $DCGL_W$,” and that “for basement structures, any residual radioactivity identified by an FSS measurement at concentrations in excess of the respective BcDCGL will be remediated.” The FCS LTP proceeds to discuss the fact that statistical sampling sizes and measurement locations may need to be adjusted to ensure that FSS scan surveys are able to adequately detect the $DCGL_{EMC}$, and the licensee will use methods consistent with MARSSIM Section 5.5.2.4, “*Determining Data Points for Small Areas of Elevated Activity*,” for that purpose. OPPD will determine the required scan MDC, which is equal to the $DCGL_{EMC}$, by calculating the area bounded by systematic measurements/samples in the survey unit and determining the area factor that corresponds to the bounded area that it calculates. OPPD will calculate the $DCGL_{EMC}$ by multiplying the area factor by the $DCGL_W$ and then comparing it to the actual scan MDC. If the actual scan MDC is less than or equal to the required scan MDC, then the spacing of the statistical systematic sampling and measurement locations is adequate to detect small areas of elevated radioactivity. If the actual scan MDC is greater than the required scan MDC, then OPPD will reduce the spacing between locations due to the lack of scanning sensitivity. The reduced area being bounded by samples will increase the applied area factor and $DCGL_{EMC}$ until it is equal to or exceeds the actual scan MDC.

FSS scan coverage is discussed in Section 5.3.4, “Scan Coverage,” of the FCS LTP, which states that the purpose of scan measurements is to confirm that OPPD properly classified the area and that any small areas of elevated radioactivity are within acceptable levels (i.e., are less than the applicable $DCGL_{EMC}$). Depending on the sensitivity of the scanning method used, OPPD may need to increase the number of total surface contamination measurement locations so the spacing between measurements is reduced. The licensee indicates that it used MARSSIM table 5.9to determine the recommended survey coverage. It also indicated that the amount of area to be covered by scan measurements is provided in table 5-20 of the FCS LTP. Investigation levels for the FSS scan surveys are discussed in Section 5.5.5.1, “Investigation Levels,” of the LTP and categorized in table 5-23 of the LTP, replicated below. Essentially, anytime the scanning response is greater than the Operational DCGL or scan MDC (if greater than the Operational DCGL) an investigation will be triggered.

Table 5-1: FSS Investigation Levels

Classification	Scan Investigation Levels	Direct Investigation Levels
Class 1	> Operational DCGL or > MDC_{scan} if MDC_{scan} is greater than Operational DCGL	> Operational $DCGL_W$
Class 2	> Operational DCGL or > MDC_{scan} if MDC_{scan} is greater than Operational DCGL	> Operational $DCGL_W$
Class 3	> Operational DCGL or > MDC_{scan} if MDC_{scan} is greater than Operational DCGL	> 0.5 Operational $DCGL_W$

5.5.3 FSS Preparation, Investigation Process, and Reclassification Activities

The licensee indicates in Section 5.3.6, "Reference Grid, Sampling, and Measurement Location," and its subsections of the FCS LTP that it will develop reference grids and systematic sampling and measurement locations in accordance with MARSSIM sections 4.8.5 and 5.5.2.5. OPPD will use a reference grid for reference purposes and to locate the sampling and measurement locations. OPPD will physically mark the reference grid during the survey to aid in the collection of samples and measurements. At a minimum, OPPD will define a benchmark for each survey unit that will serve as an origin for documenting survey efforts and results. The licensee indicates in Section 5.3.6.2, "Systematic Sampling and Measurement Locations," of the FCS LTP that reference grids and systematic sampling will follow either a triangular or square pattern with the triangular pattern being preferred in most cases.

Depending upon FSS results and the results of any investigations, there may be a need for remediation, reclassification, and resurvey of certain survey units; these concepts are described in Section 5.5.5.2, "Remediation, Reclassification and Resurvey," of the FCS LTP. The LTP states that OPPD will remediate any areas of elevated residual radioactivity above the BcDCGLs in any Class 1 survey units for media other than soil. OPPD will conduct a RASS survey to verify the effectiveness of remediation. The licensee also commits to reclassifying Class 3 survey units (or portions thereof) to Class 1 if it identifies small areas of elevated activity exceeding the Operational DCGL during FSS or through investigations. If a FSS measurement in a Class 3 survey unit exceeds 50 percent of the Operational DCGL but less than the Operational DCGL, then the survey unit, or portion of the survey unit, will be reclassified as Class 2. Similarly, if any Class 2 survey unit exceeds the operational DCGL, then it (or portions thereof) will be redesignated as a Class 1 survey unit. Any reclassification will result in OPPD redesigning the FSS strategy for the affected survey units.

The FCS LTP further states that OPPD will use DQOs to evaluate the remediation, reclassification, and/or resurvey actions it will take if an investigation level is exceeded. OPPD may reclassify a survey unit from a less restrictive classification to a more restrictive classification without prior NRC approval. However, reclassification to a less restrictive classification requires prior NRC approval. The required remediation, reclassification, and resurvey actions are further described in Table 5-24, "Remediation, Reclassification, and Resurvey Actions," of the FCS LTP.

5.5.4 NRC Evaluation of FSS Planning and Design

The NRC staff evaluated the licensee's final status survey design for FCS using the regulatory guidance and acceptance criteria contained in section 4.4 of NUREG-1757, Volume 2, Revision 2, and section 2.5 of the SRP. Based on the discussion provided in this section of the SER, the

NRC staff finds that the plans for conducting FSS are consistent with the NRC's guidance and requirements, and that the plans are adequate to assist the licensee in planning and designing the FSS for each survey unit. As such, the NRC staff finds that the licensee's FSS design methods are adequate to demonstrate, in part, compliance with 10 CFR 50.82(a)(9)(ii)(D).

5.6 Final Status Survey Methods for Residual Radioactivity Measurements

Section 5.4, "Final Status Survey Implementation," of the FCS LTP, and its subsections, describes FSS implementation and discusses the various FSS survey methods and measurements, including surface scanning, static measurements, gamma spectroscopy of volumetric materials, sampling, and in-situ gamma spectroscopy measurements. OPPD will incorporate the requirements and objectives in the LTP into procedures which will govern the surveys of relevant media (e.g., structures, surface soil, subsurface soil, excavated soil, clean fill, areas covered by asphalt or concrete, buried piping, groundwater, sediments, and surface water) and address the survey design process, survey performance, and data assessment.

5.6.1 FSS Measurement Methods for Soil

Specific sampling strategies for surface and subsurface soils are provided in Sections 5.4.1.3.1, "Sampling of Surface Soils," and 5.4.1.3.2, "Sampling Subsurface Soils," of the FCS LTP, respectively. The licensee defines surface soil as referring "to outdoor areas where the soil is, for purposes of dose modeling, considered to be uniformly contaminated from the surface down to a depth of 15 cm or 6 in. These areas will be surveyed through combinations of sampling, scanning, and *in situ* measurements." OPPD will take surface soil samples for FSS of land areas at designated systematic locations and at areas of elevated activity identified by gamma scans. The licensee commits that "if during the performance of FSS, the analysis of a surface soil sample indicates the potential presence of residual radioactivity in excess of the subsurface OpDCGLs, then additional biased soil sample(s) will be taken to the appropriate depth within the area of concern as part of the investigation." All media samples collected during FSS will be analyzed by gamma spectroscopy as well as radiochemically analyzed for C-14.

OPPD also defines subsurface soil as "soil that is greater than 15 cm below the ground surface where ground surface in this context includes the exposed surface of an excavated area before backfilling. Soil that will remain beneath structures such as basement floors/foundations or pavement at the time of license termination are considered subsurface soil." In addition, the FCS LTP notes that any soil excavation created to expose or remove a potentially contaminated subgrade basement structure will undergo FSS prior to backfill. The FSS will be designed as an open land survey using the classification of the removed structure, and the OpDCGLs for soil as the release criteria. Subsurface soil will be sampled during FSS in Class 1 open land areas at 10 percent of the systematic soil sampling locations, with the location(s) selected at random. OPPD plans to take no subsurface soil sample(s) as part of the FSS in Class 2 and Class 3 open land survey units, because the FCS HSA and site characterization activities have shown that there is minimal residual radioactivity in subsurface soil.

Sampling of subsurface soil below basement structure foundations is discussed in section 5.4.1.3.3 of the LTP, "continuing characterization will consist of soil borings or use of GeoProbe technology at the nearest locations along the foundation walls that can be feasibly accessed. The under-basement soil activity will be determined by interpreting results from borings collected at the nearest locations," and if OPPD detects residual radioactivity in subsurface soils adjacent to or under a basement surface, it investigate that residual radioactivity and "a sample plan for the investigation will be created as specified by procedure and the plan and investigation results will be provided to NRC for evaluation. Based on the results of the investigation, OPPD will assess the dose consequences of the subsurface soil contamination or will remediate, as necessary."

OPPD presented discrete radioactive particle survey considerations in section 5.4.1.11 of the LTP. The NRC staff acknowledge that OPPD's contamination control program and procedures are intended to assure it will identify and remove DRPs in a timely manner and that DRPs will not be present at the time of license termination. OPPD also stated that it would conduct adequate surveys consistent with MARSSIM guidance for the DQO process for DRPs if DRPs could be present based on-site history, remediation activities, or proximity to areas where DRPs have been previously detected.

Any excavation created to expose or remove a potentially contaminated system or component that has the potential to contain residual radioactivity in excess of the OpDCGLs will undergo FSS prior to backfill. RA surveys of excavations or of excavated soil will conform to all commitments for RAs, as described in sections 5.2.5, 5.2.6.8, 5.4.1.4, 5.4.1.5, and 8.2.1 of the LTP, including scanning, sampling or in situ measurements, and HTD ROCs analyses. The licensee notes in section 5.2.6.8, that "the average BcSOF from an RA performed in support of an excavation will be compared to the average BcSOF of the applicable open land survey unit, utilizing the surface soil BcDCGL, and the larger of the two will be included in consideration for use in the compliance matrix."

5.6.2 FSS Measurement Methods for Structures

Several sections of the LTP pertain to structural surveys and methods. Section 5.4.1.3.4 of the LTP discusses sampling of concrete, section 5.4.1.7 discusses remediation and characterization of buried piping, section 5.4.1.8 discusses embedded piping and penetrations, while section 5.4.1.9 discusses survey considerations for above-ground buildings and miscellaneous structures. Additional sections of the LTP such as section 5.4.1.2, "Fixed Measurements," and 5.4.1.1.1, "Beta-Gamma Scanning," are also applicable.

For most structures, OPPD will collect fixed measurements at random or systematic locations in above-ground building survey units, consistent with the survey unit classification, to show compliance with the release criterion. OPPD will collect fixed measurements within penetrations and embedded or buried piping survey units at pre-determined increments. OPPD will also collect fixed measurements at locations identified by scanning surveys as part of an investigation to determine the source of the elevated instrument response. OPPD will scan the surfaces of above-ground buildings and miscellaneous materials for beta/gamma radiation with instruments such as those listed in table 5-22 of the LTP. OPPD will typically perform the measurements at a distance of one-half inch or less from the surface and at a scan speed of one detector width/second for hand-held instruments. OPPD will monitor audible and visual signals for indications of elevated residual radioactivity during scanning.

While the procedures for collecting fixed measurements for basement structures are not specifically addressed in a single section of the LTP, several sections state that characterization of subsurface structures will be either through concrete sampling or in-situ gamma spec measurements. For example, Section 5.4.1.1, "Scanning," states that "for basement structures, the surface area covered by a single In-Situ Object Counting System (ISOCS) measurement is large (a nominal range of 10-30 m^2) which eliminates the need for traditional scan surveys." In Section 5.5.4 of the LTP, "Data Assessment for FSS of Basement Walls/Floors," it states that, "after a sufficient number of ISOCS measurements are taken in a basement survey unit, the data will be summarized, including any judgmental or investigation measurements. The measured activity for each gamma-emitting ROC (and any other gamma emitting radionuclide that is positively detected by ISOCS) will be recorded (in units of pCi/m^2). Background will not be subtracted from any measurement. Using the radionuclide mixture fractions applicable to the survey unit, an inferred activity will be derived for HTD ROCs using the surrogate approach." As described in the LTP, a sufficient number of ISCOCs measurements is the number necessary to meet the confidence requirements of the statistical tests, as specified in MARSSIM. When discussing scanning, the LTP defines a sufficient number as that necessary to provide 100% field of view for the detector over the entire survey unit.

Concrete sampling is discussed in section 5.4.1.3.4 of the LTP as potentially being performed in two methods. The first method is through coring using a heavy-duty drill that is bolted to a floor or wall surface. OPPD will obtain a 3-inch wide by 6-inch deep (usually) core and slice it into one-half inch pucks. OPPD will obtain deeper cores as necessary to bound the depth of contamination. The second method involves a patented procedure that uses a hollow drill bit to obtain exact volumes of concrete material at certain depths while utilizing a vacuum collection system. Material from each of the incremental depths at a location may be captured in a separate container for each depth increment via use of the vacuum system. Ten percent of concrete samples taken for continuing characterization will be analyzed by a laboratory for the initial suite of radionuclides from table 5-2. The sample(s) selected for full suite analysis to meet the 10 percent requirement will be from the highest gamma activity of the sample population; however, OPPD will send additional samples (above 10 percent) to the laboratory if the samples exhibit sufficient activity such that the HTD ROCs will likely be detectable by the laboratory using the nominal surrogate ratios and MDCs.

Section 5.4.1.7 of the LTP states that OPPD will remediate designated sections of buried piping in place as necessary. These sections will then undergo FSS. The inventory of buried piping located below 1,001 feet AMSL that will remain and undergo FSS is provided in CPP FC-21-002. OPPD will demonstrate compliance with the Operational DCGL values for buried piping, as presented in table 5-10 of the LTP, by measurements of total surface contamination. The radiological survey of pipe system interiors involves the insertion of appropriately sized detectors into the pipe interior by a simple "push-pull" methodology, whereby the OPPD can easily determine the position of the detector in the piping system in a reproducible manner. OPPD acquires a static measurement at a pre-determined interval for the areal coverage to be achieved. The measurement output represents the gamma activity in gross counts per minute (cpm) for each increment of piping traversed. This measurement value in cpm is then converted to disintegrations per minute (dpm) using the efficiency of the detector. OPPD will then adjust the total activity in dpm for the assumed total effective surface area commensurate with the pipe diameter, resulting in measurement results in units of dpm/100 cm^2. OPPD will perform radiological evaluations for piping that cannot be accessed directly via measurements made at traps and other appropriate access points where OPPD deems the radioactivity levels to either bound or be representative of the interior surface radioactivity levels.

Similar to buried piping, the list of penetrations and embedded piping to remain at license termination is provided in CPP FC-21-002. Embedded pipe and penetrations have separate OpDCGLs. The OpDCGLs for embedded piping are listed in table 5-12 of the LTP. For penetrations, OPPD will use the same DCGLs as the structure in which the penetration resides. However, the survey methods will be the same for embedded piping and penetrations. OPPD will assess and quantify the residual radioactivity remaining in each section of embedded piping/penetration applicable to each FSS unit by direct survey. OPPD will survey shallow penetrations or short lengths of embedded pipe that are directly accessible using hand-held portable detectors, such as gas-flow proportional or scintillation detectors. It will survey lengths of embedded pipe or penetrations that cannot be directly accessed by hand-held portable detectors using sodium iodide or cesium iodide detectors of appropriate size that OPPD will insert and transport through the pipe using flexible fiber-composite rods or attached to a flexible video camera/fiber-optic cable. The detector output will represent the gamma activity in gross cpm. OPPD will then convert this gamma measurement value in cpm to dpm using an efficiency factor based on the calibration source. OPPD will adjust the total activity in dpm for the assumed total effective surface area commensurate with the pipe/penetration diameter and a conservative estimate of the area of detection, resulting in measurement results in units of dpm/100 cm^2. This measurement result will then represent a commensurate and conservative gamma surface activity. OPPD will then convert the gamma surface activity for each FSS measurement to a gamma measurement result (in units of pCi/m^2) for each gamma ROC based on the mixture applicable to the pipe/sleeve surveyed. OPPD will infer HTD ROC concentrations from the applicable gamma radionuclide concentration to derive a concentration for each ROC for each measurement.

The FSS of aboveground buildings will consist of scan surveys and static measurements using a beta/gamma detector coupled to a data-logger instrument. OPPD preliminarily projects the following impacted aboveground structures to remain: Training Building; FLEX Building; OCA Access Building; Switchyard 3451 Old Building; Switchyard 3451 New Building; and 1251 Control and Switchgear Building. When performing FSS on these buildings, OPPD will use the screening values for building surface contamination from table H.1 in appendix H of NUREG-1757, Volume 2, Revision 2.

5.6.3 FSS Measurement Methods for Groundwater

Groundwater assessment is described in Section 5.4.1.10, “Groundwater,” of the FCS LTP, where OPPD notes that “Assessments of any residual radioactivity in groundwater at the site will be via groundwater monitoring wells installed at FCS. Ongoing monitoring of surface water and groundwater at FCS include REMP, Radiological Groundwater Protection Program, and National Pollutant Discharge Elimination System (NPDES) monitoring.” Additional details of groundwater assessment methods and measurements and the NRC’s evaluation of them can be found in sections 6.7 and 6.12 of this SER.

5.6.4 FSS Measurement Methods for Other Media

Additional survey strategies are discussed within section 5.4 of the FCS LTP, including those for pavement-covered areas, sediments, and surface water, and are discussed briefly in the following paragraphs.

Pavement covered areas will be incorporated into larger open land survey units and will be surveyed according to the classification of the survey units in which they are located. Surface soil DCGLs will be used for these surveys, and sample media will be pulverized and analyzed by gamma spectroscopy for comparison to the Operational DCGL. The licensee indicates in Section 5.4.1.6, “Pavement Covered Areas,” of the FCS LTP that if pavement exhibits residual radioactivity above the surface soil BcDCGL, then it will dispose of the pavement as radiological waste and perform an investigation of underlying soil.

Sediments and surface water are discussed in section 5.4.1.3.5 of the LTP. OPPD will assess sediments by collecting samples within locations of surface water ingress or by collecting composite samples of bottom sediments. OPPD will collect such samples using procedures based on accepted methods for sampling of this nature. OPPD will evaluate sediment samples against the site-specific soil OpDCGLs for each of the potential ROC as presented in table 5-8 of the LTP. OPPD will assess residual radioactivity levels in surface water drainage systems through the sampling of sediments, total surface contamination measurements, or both, at traps and other access points where OPPD expects that radioactivity levels will be representative or bounding of the residual radioactivity on the interior surfaces.

5.6.5 NRC Evaluation of FSS Methods for Measuring Residual Radioactivity

The NRC staff evaluated the licensee's plans for final status survey measurement methods for residual radioactivity using the regulatory guidance and acceptance criteria contained in section 4.4 of NUREG-1757, Volume 2, Revision 2, and section 2.5 of the SRP. Staff further evaluated radiological measurements of the media and structures discussed in sections 5.6.5.1-5.6.5.4 of this SER and found the proposed measurement methods for diffuse residual radioactivity in soil and on structural surfaces to be generally consistent with industry practice, guidance in MARSSIM and NUREG-1507, and demonstrate compliance, in part, with 10 CFR 50.82 (a)(9)(ii)(D).

5.6.5.1 NRC Evaluation of Discrete Radioactive Particles

The licensee addressed the possibility of DRPs within Section 5.4.1.11, "Survey Considerations for Suspected Discrete Radioactive Particle Areas," of the LTP. The licensee states that it will have adequate procedures suitable for detecting DRPs that may be identified or generated during decommissioning and that timely and prudent actions will be taken in response if DRPs are encountered. OPPD states that it will document information on DRPs including the radionuclides and activity, in addition to any qualitative details on DRPs (e.g., size), within the condition reporting system. OPPD will perform radiological surveys that are adequate to detect DRPs in survey units that could have DRPs based on site history, remediation activities, or proximity to areas where DRPs have been previously detected. OPPD stated that the surveys will be consistent with guidance in the MARSSIM DQO process to ensure that the survey results are of sufficient quality and quantity to support the final decision on allowing unrestricted use of the site. As such, NRC staff have reasonable assurance that DRP contamination will be adequately addressed both during decommissioning operations and during FSSs and that NRC inspectors can review the efforts made to control DRP contamination consistent with these commitments during routine inspections.

5.6.5.2 NRC Evaluation of Final Status Survey Approach for Basement Structures

The licensee utilizes MARSSIM in the development of the FSS plans for basement structures, but it deviates from the MARSSIM process in ways that warrant elaboration. A typical MARSSIM process for surfaces utilizes scanning in conjunction with random/systematic sampling (or discrete measurements). Random/systematic sampling provides a non-biased and representative approach to determine the radiological status of a survey unit, while scanning is utilized for the purpose of locating and delineating any elevated areas of activity. As such, scan surveys should generally be part of the overall FSS process. NRC staff noted that scan surveys are generally not part of the FSS process for subsurface structures at FCS, but instead appear to be incorporated into RASS surveys prior to the FSS while OPPD anticipates ISOCS measurements to satisfy the scanning requirements during FSS. In response to RAI CL-7 (ML23060A197), the licensee stated that:

"Following demolition of Class 1 structures with basements, after all debris is removed and the floors are cleaned, a RASS will be performed to ensure that any individual ISOCS measurement will not exceed the $BcDCGL_{wf}$ from table 5-5 during FSS. The survey will be performed using hand-held beta/gamma instrumentation as presented in table 5-22 in typical scanning and measurement modes.

Areas greater than the $BcDCGL_{wf}$ will be remediated and resurveyed to ensure the elevated area is successfully remediated. Areas exceeding the $OpDCGL_{wf}$ will be bounded and investigated. Investigation may include the use of ISOCS with a reduced field of view (FOV). The reduced FOVs and/or overlapping FOVs are able to identify any elevated areas."

Additional factors considered in the NRC staff's evaluation of basement surface surveys include their context within the BFM, and the currently proposed survey unit classification and investigation processes. The BFM is unique with regard to how overall residual radioactivity is assessed, which is discussed in greater detail in section 6 of this SER. With regard to current survey unit classification, all FCS basement survey units will be surveyed at a discrete measurement frequency equivalent to a Class 1 open land area, which will ensure a consistent overall survey coverage throughout the entire decommissioning process. However, staff noted that the general MARSSIM guidance for survey unit size restrictions was not being used for structures in this instance. In response to RAI TE-3 (ML23060A197), the licensee has revised table 5-1 of the LTP so that the size restrictions for basements are essentially equal to that of open land areas of the same classification. The licensee further clarifies that:

"Class 1 or Class 2 basement survey unit sizes can exceed the open land area sizes listed in Table 5-1, but the direct measurement frequency (i.e., samples per area) must be equal to or greater than that derived assuming survey unit sizes of 2,000 m^2 and 10,000 m^2, respectively. For example, if 18 direct measurements are required in a Class 1 survey unit, the sample frequency could not be less than 2,000 m^2/18 = 1 sample per 111 m^2. If the basement survey unit were 3,000 m^2 the number of direct measurements would be 3,000/111 = 27. If unit size is less than 1,000 m^2 the sample frequency increases to 1000/18 = 1 sample per 55 m^2. The same rationale would apply to a Class 2 survey unit. Regardless of the survey unit size, the scan percentage using ISOCS is as recommended in MARSSIM."

The licensee's response to RAI TE-3, combined with the previous response to perform RASS surveys involving scanning of basements prior to FSS, results in staff having reasonable assurance that adequate remediation and characterization efforts will be made to assure basements will meet the unrestricted release criteria.

With regard to investigation levels for basement surfaces, the ISOCS investigation levels (as shown in table 5-24 of the FCS LTP and in Table 5-1 of this SER for direct measurements) are based on an Operational DCGL which is a fraction of the BcDCGL as opposed to the DCGLemc. This would ensure that investigations will be triggered at lower levels than the BcDCGL (the licensee commits to remediating any residual radioactivity greater than the BcDCGLs in basements), but the NRC staff notes that if ISOCS surveys were performed for the sole purpose of locating small areas of elevated activity (as is the case using a typical MARSSIM scanning model), investigations may need to be designed differently.

As a point of reference, the NRC staff notes that an NRC-sponsored study was completed in 2006 titled “Spatially-Dependent Measurements of Surface and Near-Surface Radioactive Material Using In-situ Gamma Ray Spectrometry (ISGRS) For Final Status Surveys” (ML17284A121).This study addressed various survey and investigation considerations when a discrete particle is located within a larger in situ FOV, and states the following with respect to investigation levels:

> “It is important to understand that scanning is performed to identify or detect the presence of areas of elevated contamination, which may be discrete particles. The purpose of scanning is not to quantify the activity in the elevated area. The difference is one of detectability versus measurability. According to MARSSIM, “Scanning surveys are performed to locate radiation anomalies indicating residual gross activity that may require further investigation or action” (MARSSIM 2000). Therefore, it is necessary to define ISGRS investigation levels during scanning, and to specify the nature of the further investigation once the investigation level is triggered.”

The 2006 ISGRS study goes on to note that if the investigation level is exceeded, then conventional scan surveys might be conducted to confirm and/or identify the location of the discrete particle. The FCS LTP does not specify that conventional scan surveys will be performed in the event that an ISOCS investigation level is exceeded, but rather seems to rely on preliminary demolition and RASS surveys to locate areas of elevated activity.

Based on its evaluation of the entire survey process for basement structures, the NRC staff finds the overall survey ISOCS methodology, when supplemented by the licensee’s commitment to perform pre-FSS scanning surveys (i.e., RASS surveys) and its commitment to remediate any elevated residual radioactivity greater than the BcDCGLs, consistent with the MARSSIM guidance regarding the radiological sampling approach and detection of elevated areas. The ISOCS detection sensitivities being much less than the ETD DCGLs, combined with the surrogate ROC process or inferring of HTD ROC concentrations, should be adequate to quantify the average residual radioactivity within its field of view. The licensee’s process for developing a sum of fractions result from basement surfaces based on the ISOCS measurements should be adequate to demonstrate compliance with the DCGLs and also demonstrates compliance, in part, with 10 CFR 50.82 (a)(9)(ii)(D).

5.6.5.3 NRC Evaluation of Final Status Survey Approach for Buried Piping, Embedded Piping, and Penetrations

The NRC staff evaluated the licensee’s plans of considering buried piping at FSS, as described in sections 5.4.1.7 and 5.4.1.8 of the FCS LTP, and as previously discussed in the SER. The survey methods and instrumentation proposed for buried piping surveys, including the usage of handheld portable detectors and push-pull methods, provide reasonable assurance that the buried piping can be sampled to an extent that will allow the licensee to demonstrate compliance with the unrestricted release dose criteria of 10 CFR 20.1402. In addition, the proposed strategies to maintain adequate survey coverage within piping provide reasonable assurance that the survey will be consistent with the MARSSIM guidance, and NUREG-1757 guidance associated with buried piping. As such, the buried piping, embedded piping, and penetration survey methodologies are considered adequate because they are consistent with MARSSIM and NUREG-1757, Volume 2, Revision 2 and are demonstrate compliance, in part, with 10 CFR 50.82 (a)(9)(ii)(D).

5.6.5.4 NRC Evaluation of Final Status Survey Approach for Open Land Survey Areas and Soils

The NRC staff evaluated the licensee's final status survey approach for open land survey units, as provided in section 5.4.1.3.1 of the FCS LTP, and as previously discussed in the SER. The NRC staff determined that the FSS design and usage of the Data Quality Objective (DQO) and Data Quality Assessment (DQA) processes is consistent with MARSSIM and NUREG-1757, Volume 2, Revision 2. The proposed survey instrumentation and detection sensitivities are considered adequate to detect the ETD ROCs at levels below the DCGLs. The scan coverage and systematic sampling strategies for open land areas are consistent with MARSSIM and NUREG-1757 and are therefore considered adequate. The sum of fractions and elevated measurements comparison methods, as presented in Equation 5-9 of the FCS LTP, is consistent with MARSSIM and NUREG-1757, Volume 2, Revision 2. For these reasons, the NRC staff determined the survey approach for land survey areas to be adequate and to demonstrate compliance, in part, with 10 CFR 50.82 (a)(9)(ii)(D).

5.6.5.5 NRC Evaluation of Final Status Survey Approach for Remaining Above Ground Structures

The NRC staff evaluated the licensee's final status survey approach for impacted aboveground buildings, structures, and equipment that will remain after license termination, as provided in section 5.4.1.9 of the FCS LTP, and as previously discussed in the SER. The licensee states that the "FSS of minor solid structures, such as but not limited to the switchyard, telephone poles, fencing, culverts, duct banks and electrical conduit will be included in the FSS of the open land survey unit in which they reside" so NRC staff expect those surveys to be addressed in the FSSRs for the open land areas. The NRC staff determined that OPPD's FSS design, which uses the screening values for building surface contamination from table H.1 of NUREG-1757, Volume 2, Revision 2 is generally consistent with NRC guidance.

One potentially problematic method for building surveys was noted by the NRC staff in that one of the licensee's ambient background determination methods could be non-conservative (i.e., laying a detector on its side on the surface of concern to obtain ambient background readings). While other methods discussed in the LTP appear appropriate, such as obtaining background from non-impacted materials, the staff notes that a non-conservative bias in measurement methods may occur as a result of this particular method if it is utilized without proper precautions being taken such as verifying no hot spots of residual radioactivity being nearby. If NRC staff note a significant non-conservative bias during verification or confirmatory surveys, the licensee may need to either correct its data or omit utilizing this particular method for determining ambient background. Otherwise, the NRC staff found the approach taken by the licensee for aboveground structures to be consistent with guidance in MARSSIM and NUREG-1757, Volume 2, Revision 2. As such, staff find the FSS methods for aboveground structures to be adequate and demonstrate compliance, in part, with 10 CFR 50.82 (a)(9)(ii)(D).

5.7 Final Status Survey Instrumentation

5.7.1 FSS Instrument Selection, Calibration, and Sensitivity

Section 5.4.2, "Final Status Survey Instrumentation," of the FCS LTP discusses instrument selection, which is based on detection sensitivity, operating characteristics and expected performance in the field. This section states that OPPD will use the DQO process in selecting FSS instruments. With regard to instrument detection capability, the licensee notes that “detection and measurement instrumentation for performing FSS is selected to provide both reliable operation and adequate sensitivity to detect the ROC identified at the site at levels sufficiently below the Operational DCGL,” and “specific DQOs for instruments are established early in the planning phase for FSS activities, implemented by standard operating procedures (SOP) and executed in the survey plan.” Section 5.4.2.1, “Instrument Selection,” of the FCS LTP additionally notes that for “direct measurements and sample analyses, MDCs less than 10% of the Operational DCGL are preferable while MDCs up to 50% of the Operational DCGL are acceptable.” Section 5.4.2.1. of the LTP also states that “instruments used for scan measurements in Class 1 areas will be required to be capable of detecting radioactive material at the Base Case DCGL”.

The licensee’s proposed FSS instrumentation is listed in table 5-22 of the LTP, and typical instrument MDCs are presented in Table 5-23 of the LTP, “Typical FSS Instrument Detection Sensitivities.” The licensee also indicates in section 5.4.2.1 of the LTP that “other measurement instruments or techniques may be utilized,” and “acceptability of additional or alternate instruments or technologies for use in the FSS will be justified in a technical basis evaluation document prior to use.” The licensee commits to developing technical basis evaluations for alternate final status survey instruments or techniques that will be provided for NRC review 30 days prior to use. The evaluation contained in the technical basis document will include the following:

- a description of the conditions under which the alternate method would be used;
- a description of the measurement method, instrumentation, and criteria;
- a justification that the alternative technique would provide the required sensitivity for the given survey unit classification; and,
- a demonstration that the instrument provides sufficient sensitivity for measurement.

The licensee notes that the measurement sensitivity (i.e., MDC) is dependent upon the counting time, geometry, sample size, detector efficiency and background count rate. OPPD describes the scan and static MDC calculations for FSS which utilize guidance from NUREG-1507. Scan MDC calculations for gamma scans of land areas are described in Section 5.4.2.4.4 of the LTP, “Gamma Scan Minimum Detectable Concentration,” along with a basic description of the scanning procedures that OPPD will utilize for gamma walkover surveys of land areas. This section states that the methods used for surface scanning of land areas will be consistent with NUREG-1507 and use essentially the same geometry configuration as the model used in MARSSIM for calculating the sensitivity.

The calibration of instrumentation used for FSS is discussed in Section 5.4.2.2, “Calibration and Maintenance,” of the FCS LTP, where it is stated that “Radioactive sources used for calibration will be traceable to the NIST and have been obtained in standard geometries to match the type of samples being counted.” OPPD will perform instrument calibrations for the radiation types and energies of interest or to a conservative energy source with calibration certificates and/or forms maintained with the instrumentation and project records. Instrument response checks and measurement sensitivity are described in Section 5.4.2.3, “Response Check.” Prior to daily use, OPPD will verify all project instruments, and initial response data collected and again at the end of use.

Section 5.4.2.4.5, "[High Purity Germanium] HPGe Spectrometer Analysis," and Section 5.4.2.4.6, "Pipe Survey Instrumentation," of the FCS LTP address HPGe spectrometer analysis and pipe survey instrumentation, respectively. Regarding HPGe analysis, the licensee indicates that the "onsite FCS laboratory maintains gamma isotopic spectrometers that are calibrated to various sample geometries, including a 500 mL flat-bottomed beaker geometry for soil analysis using Canberra LABSOCS software," and "systems are calibrated using a NIST-traceable mixed gamma source." With regard to pipe survey instrumentation, the licensee states detection sensitivities range from approximately 350 dpm/100 cm^2 to 5,200 dpm/100 cm^2, and that this level of sensitivity is adequate to detect residual radioactivity below the OpDCGLs derived for the unrestricted release of buried pipe.

5.7.2 NRC Evaluation of Final Status Survey Instrumentation

The NRC staff evaluated the licensee's proposed radiation detection and measurement instrumentation for performing FSS with the regulatory guidance and acceptance criteria contained in section 4.4 of NUREG-1757, Volume 2, Revision 2, and section 2.5 of the SRP. The NRC staff attempted to verify the static and scan MDCs for the Ludlum 44-116 in the table using the data and equations in Subsections of 5.4.2.1 and 5.4.2.4 the LTP. The MDCs for this instrument were not reproducible. The NRC staff calculated MDCs, although higher than those in the LTP, remain less than the DCGLs, thus, this does not pose a risk significant issue. In addition, the column labeled "Typical Instrument Efficiency$_b$ ($\mathcal{E}_t$)" in the table is representative of $\mathcal{E}_i$ rather than $\mathcal{E}_t$. These discrepancies were communicated to the licensee, who indicated the values in the table were meant to be typical, not precise, which the NRC staff accepts because precise instrument parameters are not a reasonable commitment when fluctuation in the parameter values is expected among instruments. Also, OPPDs states that new technologies will be documented in a technical basis document provided to the NRC at least 30 days prior to use. Based on the discussion provided in this section of the SER and the communications noted in this paragraph, the NRC staff finds that the proposed radiation detection and measurement instrumentation for performing FSS are consistent with MARSSIM, NUREG-1575, and NURG-1507. As such, the NRC staff finds that the licensee's proposed FSS instrumentation adequate to demonstrate compliance with 10 CFR 50.82(a)(9)(ii)(D).

5.8 Quality Assurance

5.8.1 FCS Decommissioning Quality Assurance Project Plan

OPPD's QA program for decommissioning and completion of license termination activities at FCS is described in Section 5.6, "Quality Assurance," of the LTP and its subsections.
Project management and the FCS decommissioning organizational structure are described in Section 5.6.1, "Project Management and Organization," of the FCS LTP, and the licensee notes that further details on key positions are described in the project Quality Assurance Project Plan (QAPP) (ML21271A173). The basic elements of the QAPP are described in Section 5.6.2, "Quality Objectives and Measurement Criteria," of the FCS LTP, and include: written procedures; training and qualifications; measurement and data acquisitions; instrument selection, calibration, and operation; chain of custody; control of consumables; control of vendor-supplied services; database control; and data management. The licensee states that the QA objectives for FSS are to ensure the survey data collected is of the type and quality needed to demonstrate, with sufficient confidence, that the site is suitable for unrestricted release. OPPD said this objective is met through use of the DQO process for FSS design, analysis, and evaluation. OPPD also said that compliance with the QAPP ensures that the following items are accomplished: (1) the elements of the FSS plan are implemented in accordance with the

approved procedures; (2) surveys are conducted by trained personnel using calibrated instrumentation; (3) the quality of the data collected is adequate; (4) all phases of package design and survey are properly reviewed, with QC and management oversight provided; and (5) corrective actions, when identified, are implemented in a timely manner and are determined to be effective.

Measurement and data acquisition is described in Section 5.6.3, "Measurement/Data Acquisition," and its subsections of the LTP, and the licensee lists the following QC measures for use during decommissioning: replicated measurements and surveys; duplicate and split samples; field blanks and spiked samples; and QC investigations. OPPD will perform replicate measurements, duplicate and split sampling on an approximate 5% frequency while replicate surveys will be performed as directed by the LT/FSS Manager. The acceptance criteria of replicate measurements and surveys is whether the same conclusions from the original and replicate data are reached. The acceptance criteria of duplicate and spit sampling results will be determined consistent with MARLAP (NUREG-1576), section C.4.2.2. The licensee's QA assessment and oversight strategies are described in Section 5.6.4, "Assessment and Oversight," and its subsections of the LTP, which includes focused self-assessments by project management; independent review of survey results; and the use of a sitewide corrective action process.

5.8.2 NRC Evaluation of Quality Assurance

The NRC staff evaluated the licensee's QA program for the decommissioning of FCS using the regulatory guidance and acceptance criteria contained in section 4.4.5 of NUREG-1757, Volume 2, Revision 2, and section 2.5 of NUREG-1700. Based on the discussion provided throughout the LTP and discussed in this section of the SER, the NRC staff finds that the FCS QA program: assures support of both field survey work and laboratory analysis; addresses the organization responsible for QA; addresses training and qualification requirements; addresses survey instructions and procedures, including water, air, and soil sampling procedures; addresses document control; addresses inspections; addresses control of survey equipment; addresses handling, storage, calibration, and response checks; addresses shipping of survey equipment and laboratory samples; addresses nonconformance items; addresses corrective action; addresses QA records; and addresses the survey audits, including methods to be used for reviewing, analyzing, and auditing data. The NRC staff finds that the licensee's QA program is consistent with section 2.5 of NUREG-1700 and, therefore, adequate to demonstrate compliance with 10 CFR 50.82(a)(9)(ii)(D).

5.9 Final Status Survey Data Assessment

5.9.1 FSS Data Assessment and Validation using the DQA and DQO Process

FSS data assessment is described in Section 5.5, "Final Status Survey Data Assessment," and its associated subsections of the LTP, which states that the DQA approach OPPD is implementing at FCS is an evaluation method used during the assessment phase of FSS to ensure the validity of FSS results and demonstrate achievement of the survey plan objectives. The level of effort OPPD expends during the DQA process will typically be consistent with the graded approach used during the DQO process. The DQA process includes a review of the DQOs, survey plan design, and preliminary data. It will use appropriate statistical testing, verify the assumptions of the statistical tests, and draw conclusions from the data. The DQA process includes:

- Verification that the measurements were obtained using approved methods.
- Verification that the quality requirements were met.
- Verification that the appropriate corrections were made to any gross measurements and that the data is expressed in the correct reporting units.
- Verification that the MDC requirements have been met.
- Verification that the measurements required by the survey design, and any measurements required to support investigation(s) have been included.
- Verification that the classification and associated survey unit design remain appropriate based on a preliminary review of the data.
- Subjecting the measurement results to the appropriate statistical tests.
- Determining if the residual radioactivity levels in the survey unit meet the applicable release criterion and if any areas of elevated radioactivity exist.

OPPD indicates that once the FSS data is collected, it will assess and evaluate the information for each survey unit to ensure that it is adequate to support the release of the survey unit. Simple assessment methods such as comparing the survey data mean result to the appropriate OpDCGL will be performed. OPPD will calculate the SOF as several radioisotopes are measured and, if any measurements in the data set exceed unity, it will use a nonparametric statistical test (i.e., the Sign Test) to evaluate the final dataset to determine if the data are consistent with meeting the release criteria. OPPD will investigate measurements exceeding investigation action levels. In Class 1 soil areas, OPPD will either evaluate exceedances of the OpDCGLs against the elevated measurement criteria or remediate and re-survey the locations. Once the assessment and evaluation are complete, conclusions will be made as to whether the survey unit meets the site release criteria or whether additional actions will be required.

Data validation is discussed in Section 5.5.2.1, “Data Validation,” of the FCS LTP, and the licensee notes that, at a minimum, the following actions should be included:

- Ensure that the instrumentation MDC for direct measurements and sample analyses was less than 10 percent of the Operational DCGL, which is preferable. MDCs up to 50 percent of the Operational DCGL are acceptable.
- Ensure that the instrument calibration was current and traceable to NIST standards.
- Ensure that the field instruments used for FSS were source checked with satisfactory results before and after use each day that data were collected.
- Ensure that the MDCs and assumptions used to develop them were appropriate for the instruments and techniques used to perform the survey.
- Ensure that the survey methods used to collect data were proper for the types of radiation involved and for the media being surveyed.

- Ensure that the sample was controlled from the point of sample collection to the point of obtaining results.

- Ensure that the data set is comprised of qualified measurement results collected in accordance with the survey design which accurately reflect the radiological status of the facility.

- Ensure that the data have been properly recorded.

Graphical data analyses are discussed in Section 5.5.2.2, “Graphical Data Review,” of the FCS LTP and will include, at a minimum, posting plots of measurement locations or data on maps of the site and/or surface and creating frequency plots or histograms OPPD notes that additional data review methodologies are found in MARSSIM Section 8.2.2, “Conduct a Preliminary Data Review” and may also be used.

5.9.2 FSS Statistical Tests and Data Conclusions

The FSS statistical test (i.e., Sign Test) is discussed in Section 5.5.3 “Applying Statistical Test,” of the LTP. The licensee notes that it will apply the SOF or unity rule to FSS data as discussed in the guidance contained in NUREG-1757, Volume 2, Revision 2, Section 2.7, “Sum of Fractions.” OPPD will base the sum of fractions calculation on the OpDCGLs, and if a surrogate for HTD ROC is considered in the dataset, OPPD will calculate the “unity rule equivalents” using the surrogate adjusted Operational DCGL as shown by the following equation demonstrating Cs-137 as a surrogate radionuclide for Co-60 (Equation 5-22 of the LTP, replicated below).

Sum of Fractions Calculation for Surrogate Radionuclides

$$SOF \leq 1 = \frac{Conc_{Cs\text{-}137}}{DCGL_{Cs\text{-}137_s}} + \frac{Conc_{Co\text{-}60}}{DCGL_{Co\text{-}60}} + \ldots \frac{Conc_n}{DCGL_n}$$

where:

$Conc_{Cs\text{-}137}$ = Measured mean concentration for Cs-137
$DCGL_{Cs\text{-}137s}$ = Surrogate Operational DCGL for Cs-137
$Conc_{Co\text{-}60}$ = Measured mean concentration for Co-60
$DCGL_{Co\text{-}60}$ = Operational DCGL for Co-60
$Conc_n$ = Measured mean concentration for radionuclide n
$DCGL_n$ = Operational DCGL for radionuclide n

The process for an elevated measurement comparison (EMC) is discussed in Section 5.5.3.3, “Elevated Measurement Comparison,” of the LTP. As previously noted, EMC is only applicable to Class 1 open land survey units. The $DCGL_{EMC}$, which is a BcDCGL modified by an area factor to account for small areas of elevated activity, will be used as discussed in MARSSIM Section 8.5.1, “Elevated Measurement Comparison,” and Section 8.5.2, “Interpretation of Statistical Test Results.” This analysis will use the BcDCGLs as presented in the following equation (Equation 5-23 of the LTP, replicated below).

EMC Evaluation Calculation

$$\frac{\delta}{DCGL_W} + \frac{\tau_1\text{-}\delta}{DCGL_{EMC_1}} + \frac{\tau_2\text{-}\delta}{DCGL_{EMC_2}} + \cdots \frac{\tau_n\text{-}\delta}{DCGL_{EMC_n}} < 1$$

where:

δ	= The survey unit average activity
$DCGL_W$	= The survey unit BcDCGL concentration
τ_n	= The average activity value of hot spot n
$DCGL_{EMCn}$	= The $DCGL_{EMC}$ concentration of hot spot n

OPPD stated that the fractions for all of the terms (including all elevated areas within a survey unit) will be summed and must be less than unity for the survey unit to pass the EMC evaluation.

Section 5.5.6, “Data Conclusions,” of the LTP states that the results of the statistical testing, including the application of the EMC, allow for one of two conclusions to be made. The first conclusion is that the survey unit meets the site release criterion, and the data provide statistically significant evidence that the level of residual radioactivity within the survey unit does not exceed the release criteria. OPPD may then decide to release the survey unit with sufficient confidence and without any further analyses by OPPD. The second conclusion that can be made is that the survey unit fails to meet the release criteria. The unit could fail to meet the criteria because the data may not be conclusive in showing that the residual radioactivity is less than the release criteria. In that instance, OPPD would analyze the data to determine the reason for failure, and evaluate whether the number of measurements made and the standard deviation of the measurement data are adequate to ensure that the power of the statistical tests is sufficient. It can then determine whether the issue is insufficient power or whether more remediation is necessary. Potential reasons for failing to meet the release criteria may include:

- The average residual radioactivity exceeds the Operational DCGL.

- The average residual radioactivity in soil is less than the BcDCGL; however, the survey unit fails the EMC test.

- The survey design or implementation was insufficient to demonstrate compliance for unrestricted release, (i.e., an adequate number of measurements was not performed).

- The test did not have sufficient power to reject the null hypothesis (i.e., the result is due to random statistical fluctuation).

5.9.3 Data Assessment of Basement Walls/Floors

A discussion on data assessment for basement surface FSS results is provided in Section 5.5.4, “Data Assessment for FSS of Basement Walls/Floors,” of the LTP. The licensee plans to take ISOCS measurements in each basement survey unit. The licensee will utilize a SOF approach for each measurement by dividing the reported concentration of each ROC by the Operational DCGL for basement structures ($OpDCGL_B$)for each ROC, in order to calculate an individual ROC fraction which is then summed to derive a total SOF for each measurement. OPPD will base the SOF for gamma-emitting ROCs on the measured activity and will infer the SOF for HTD ROCs utilizing the surrogate approach from Section 5.2.6.2, “Surrogate Radionuclides,” of the LTP. OPPD will not subtract background from any measurement.

The LTP also notes that OPPD will use the Sign Test to evaluate the remaining residual radioactivity against the compliance dose criterion, and will use the SOF for each measurement as the sum value for the Sign Test. The licensee further notes that “if the Sign Test demonstrates that the mean activity for each ROC is less than the $OpDCGL_B$ at a Type 1 decision error of 0.05, then the mean of all the total SOFs for each measurement in a given survey unit is calculated,” and that “if the Sign Test fails, or if the mean of the total SOFs in a basement exceeds one (using OpDCGLs), then the survey unit will fail FSS.”

OPPD will define areas of elevated activity in basement walls/floors as any area identified by measurement/sample (systematic or judgmental) that exceeds the Operational DCGL but is less than the BcDCGL. OPPD will remediate any area that exceeds the BcDCGL will be remediated. The SOF (based on the Operational DCGL) for a systematic or a judgmental measurement/sample(s) may exceed one without remediation as long as the survey unit passes the Sign test and the mean SOF (based on the Operational DCGL) for the survey unit does not exceed one. Once the survey data set passes the Sign Test (using OpDCGLs), OPPD will use the mean radionuclide activity (pCi/m^2) for each ROC from systematic measurements along with any identified elevated areas from systematic and judgmental samples with the BcDCGLs to perform a SOF calculation for each surface FSS unit in a basement in accordance with the equations presented in section 5.9.2 of this SER. The dose from residual radioactivity assigned to the FSS unit is calculated using Equation 5-8 in the LTP.

5.9.4 NRC Evaluation of Final Status Survey Data Assessment

The NRC staff evaluated the licensee’s FSS data assessment methods using the regulatory guidance and acceptance criteria contained in section 4.4 of NUREG-1757, Volume 2, Revision 2, and section 2.5 of the SRP. Based on the discussion provided in this section of the SER, the NRC staff finds that the FCS FSS data assessment methods are consistent with the guidance in MARSSIM and NUREG-1757, Volume 2, Revision 2. As such, the NRC staff finds that the FSS data assessment methods are consistent with guidance and adequate to demonstrate compliance with 10 CFR 50.82(a)(9)(ii)(D).

5.10 Final Radiation Survey Reporting

Section 5.7, “Final Radiation Survey Reporting,” of the FCS LTP discusses final radiation survey reporting, and Section 5.7.1, “FSS Unit Release Records,” and Section 5.7.2, “FSS Final Reports,” of the FCS LTP discuss survey unit release records and final FSS reports, respectively. The licensee indicates that these reports will be consistent with MARSSIM Section 8.6, “Documentation,” and that it will prepare an FSS release record to provide a complete record of the as-left radiological status of an individual survey unit, relative to the specified release criteria, while it will prepare an FSS final report to provide a summary of the survey results and the overall conclusions which demonstrate that the site, or portions of the site, meets the radiological criteria for unrestricted use, including ALARA.

5.10.1 FSS Release Records

In section 5.7.1 of the FCS LTP, the licensee indicates that the FCS FSS survey unit release records will include the following:

- Survey unit description, including unit size, descriptive maps, plots or photographs and reference coordinates

- Classification basis, including significant HSA and characterization data used to establish the final classification

- DQOs stating the primary objective of the survey

- Survey design describing the design process, including methods used to determine the number of samples or measurements required based on statistical design, the number of biased or judgmental samples or measurements selected and the basis, method of sample or measurement locating, and a table providing a synopsis of the survey design

- Survey implementation describing survey methods and instrumentation used, accessibility restrictions to sample or measurement locations, number of actual samples or measurements taken, documentation activities, QC requirements and scan coverage

- Survey results including types of analyses performed, types of statistical tests performed, surrogate ratios, statement of pass or failure of the statistical test(s)

- Survey results for RA and RASS, if performed

- QC results to include discussion of split samples and/or QC replicate measurements

- Results of any investigations

- Any remediation activities, both historic and resulting from the performance of the FSS

- Any changes from the FSS survey design including field changes

- DQA conclusions

- Any anomalies encountered during performance of the survey or in the sample results

- Conclusion as to whether or not the survey unit satisfied the release criteria and whether or not sufficient power for the statistical tests was achieved

5.10.2 FSS Final Reports

Section 5.7.2 of the FCS LTP notes that FSS final reports will be written, to the extent practical, as stand-alone documents that will usually incorporate multiple survey unit release records and may be submitted in a phased approach. The licensee commits in section 5.7.2 of the FCS LTP to include the following information in these FSS final reports:

- A brief overview discussion of the FSS program, including descriptions regarding survey planning, survey design, survey implementation, survey data assessment, and QA and QC measures.

- A description of the site, the applicable survey area(s) and survey unit(s), a summary of the applicable HSA information, conditions at the time of survey, identification of potential contaminants, and radiological release criteria.

- A discussion regarding the DQOs, survey unit designation and classification, background determination, FSS plans, survey design input values and method for determining sample size, instrumentation (detector efficiencies, detector sensitivities, instrument maintenance and control and instrument calibration), ISOCS efficiency calibration geometry, survey methodology, QC surveys, and a discussion of any deviations during the performance of the FSS from what was described in the LTP.

- A description of the survey findings including a description of surface conditions, data conversion, survey data verification and validation, evaluation of number of sample and measurement locations, a map or drawing showing the reference system and random start systematic sample locations, and comparison of findings with the appropriate Operational DCGL or action level, including statistical evaluations.

- Description of any judgmental and miscellaneous sample data collected in addition to those required for performing the statistical evaluation.

- Description of anomalous data, including any areas of elevated direct radiation detected during scanning that exceeded the investigation level or measurement locations in excess of the Operational DCGL.

- If a survey unit fails the statistical test, a description of any changes in initial survey unit assumptions relative to the extent of residual radioactivity, the investigation conducted to ascertain the reason for the failure, and the impact that the failure has on the conclusion that the facility is ready for final radiological surveys, as well as a discussion of the impact of the failure on survey design and results for other survey units.

- Description of how good housekeeping and ALARA practices were employed to achieve final activity levels.

- Within the last submitted FSS Final Report, the final dose compliance equation will be provided that lists the maximum dose to each of the dose components as described in section 5.2.6.8 and Equation 5-5 of the LTP. This information may be submitted as an attachment to a letter requesting partial site release.

5.10.3 Area Surveillance Following Final Status Survey

The licensee discusses surveillance following the FSS in Section 5.2.3.2, "Area Surveillance Following FSS," of the FCS LTP, and notes that it will implement isolation and control measures in accordance with approved FCS site procedures which will remain in force until there is minimal risk of recontamination from decommissioning, or the survey unit has been released from the license. Section 5.2.3.2 of the FCS LTP also notes that OPPD will perform documented surveillances of open land survey units on a semi-annual basis to provide additional assurance that the survey units that have successfully undergone FSS remain unchanged until final site release. The licensee indicated that the routine surveillance activities will consist of the following:

- Review of access control entries since the performance of FSS or the last surveillance,

- A walk-down of the areas to check for proper barricades and posting, as applicable, and

- A check for materials introduced into the area or any disturbance that could change the as-left radiological or physical conditions encountered during FSS, including the potential for contamination from adjacent decommissioning activities.

If evidence is found of materials that have been introduced into the survey unit or any disturbance that could change the FSS results, the licensee will perform and document judgmental scans and sampling of the survey unit, focusing on access and egress points and any areas of disturbance or concern. If the results of the survey indicate that any direct measurement (in the instance of buildings or piping) or sample result (for land survey units) is statistically greater than the initial FSS results (that is, the result is > 2 standard deviations from the initial FSS mean), then OPPD will conduct an investigation survey (in the form of an RA) of the area. If the results of the RA are statistically different than the original FSS results (that is, the result is > 2 standard deviations from the initial FSS mean), then OPPD will perform a full FSS of the affected survey unit in accordance with the LTP.

NRC staff were initially unsure of the licensee's strategy for isolation and control after conducting FSS of areas and issued a RAI, TE-4 (ML23060A197), to which the licensee responded, in part, that:

> "Physical isolation and control (I&C) measures such as ropes, gates, jersey barriers, or doors will be used to ensure that no activities that could potentially contaminate a survey unit once a decision has been made to transition that survey unit to isolation. These physical I&C measures are implemented for all open land or structure survey units when they are ready for FSS, with the exception of survey units where it is not practical to construct a barrier (e.g., large Class 3 open land survey units with no close proximity to major D&D activities or shared land areas such as farmland or state park land).
>
> In cases of survey units where a physical barrier is impractical and there is a low risk for cross-contamination, alternate I&C measures will be implemented. These include the use of signs that require contact with the LT/FSS group if entry is requested. Personnel wishing to enter an LT/FSS I&C area should contact the LT/FSS group, sign in/sign out on an I&C log, and undergo the process of frisking of personnel and equipment in and out of the area. Exemptions for certain processes can be given to people who pose little to no risk for contaminating a controlled area (e.g., state park workers, farmers). A surveillance is not relied upon as an I&C measure, rather it is an activity performed to verify that I&C measures are being maintained.
>
> Movement of materials across survey units, e.g., regrading of the site, that are under I&C would be allowed to occur prior to the release of the site, given that the proper protocols are followed regarding surveillance. Final site grading will not be performed until after license termination.
>
> Investigation surveys (e.g., RAs or surveillances) may be used for justification that no reperformance of an FSS is necessary. The documentation of such surveys does not directly supplement previously performed FSSs...the surveillance survey documentation is available to NRC for inspection and can be provided to the NRC upon request. If there is something discovered during a surveillance that challenges the data reported in the survey unit release record, FSS in that survey unit is reperformed, and a revision to the existing release record is prepared."

5.10.4 NRC Staff Evaluation of Final Radiation Survey Reporting and Surveillance following FSS

The NRC staff evaluated chapter 5, section 5.7 and its associated subsections of the FCS LTP to review OPPD's proposed documentation strategy and surveillance strategies. In conducting this review, the NRC staff used guidance in NUREG-1757, Volume 2, Revision 2, and MARSSIM. NRC staff found that OPPD's commitments in the LTP for FSS Reporting and Surveillance, as well as control of the areas once the FSS is complete, will be consistent with MARSSIM and NUREG-1757, Volume 2, Revision 2. Therefore, the planned Final Radiation Survey Report adequately demonstrates compliance, in part, with 10 CFR 50.82 (a)(9)(ii)(D).

5.11 NRC Conclusion on the Licensee's Final Status Survey Plan

The NRC staff evaluated OPPD's final status survey plans using the regulatory guidance and evaluation criteria contained in section 4.4 of NUREG-1757, Volume 2, Revision 2, and section 2.5 of NUREG-1700. Based on the discussion in the LTP and response to RAIs as cited in the evaluations of specific subsections in this SER, the staff finds that the plans provide reasonable assurance that the licensee will be able to perform adequate surveys, if performed consistent with the LTP and barring unforeseen obstacles, to demonstrate compliance with the radiological criteria for unrestricted use, as specified in 10 CFR 20.1402 and also with 10 CFR 20.1501 (a) and (b). Additionally, the staff finds that OPPD's final status survey plans are adequate to demonstrate compliance with 10 CFR 50.82(a)(9)(ii)(D).

6 COMPLIANCE WITH RADIOLOGICAL CRITERIA FOR LICENSE TERMINATION

Subpart E to 10 CFR Part 20, "Radiological Criteria for License Termination," establishes criteria for the release of sites for unrestricted use. Specifically, under 10 CFR 20.1402, the residual radioactivity that is distinguishable from background levels must result in a TEDE to the average member of the critical group that does not exceed 0.25 mSv/yr (25 mrem/yr) and the residual radioactivity must also be reduced to levels that are ALARA. Details related to the application of ALARA are discussed in chapters 4 and 5. Chapter 6, "Compliance with the Radiological Criteria for License Termination," of the FCS LTP describes the dose modeling and calculations used to establish the site-specific DCGLs that OPPD will apply to the FCS site during final status surveys in order to demonstrate compliance with the radiological criteria for release for unrestricted use contained in 10 CFR 20.1402. Staff reviewed this information using Section 2.6 of NUREG-1700, "Compliance with the Radiological Criteria for License Termination," which refers to multiple sections in NUREG-1757, Volume 2, Revision 2, for additional details. In addition, the NRC staff is assessing the licensee's compliance with the requirements set forth in 10 CFR 50.82(a)(9)(ii)(D), which requires the LTP to include detailed plans for the final radiation survey, because the DCGLs are used in the final radiation surveys. As noted in section 1.3 of this SER, the NRC's review of the dose assessment and DCGL development is predicated on the site conditions being as described in the LTP.

6.1 Approach for Overall Dose Compliance

The potentially contaminated media expected to remain on site at the time of license termination includes soil, basements (including basement walls floors, embedded piping, and fill material), buried pipe, above-ground buildings, and groundwater. To demonstrate that the overall combined dose at the site from all sources at the time of license termination is consistent with the criteria for unrestricted use in 10 CFR 20.1402, the licensee will apply the below equations (reproduced from Equations 5-6/6-26 and 5-7 in the LTP).

Equation 6-1: Compliance Dose Equation (Equations 5-6 and 6-26 in the FCS LTP)

$D_C = D_{b,w\,f} + D_s + D_{b,ep} + D_{b,f} + D_{bp} + D_{agb} + D_{e\,gw}$

where:

D_C	=	compliance dose (mrem/yr)
$D_{b,wf}$	=	maximum survey unit dose from basement walls and floors (mrem/yr)
D_s	=	maximum survey unit dose from soil (mrem/yr)
$D_{b,ep}$	=	maximum survey unit dose from basement embedded pipe (mrem/yr)
$D_{b,f}$	=	maximum survey unit dose from basement fill (mrem/yr)
D_{bp}	=	maximum survey unit dose from buried pipe (mrem/yr)
D_{agb}	=	maximum survey unit dose from above-ground buildings (mrem/yr)
D_{egw}	=	maximum dose from existing groundwater (mrem/yr)

Equation 6-2: Compliance Dose Equation (Equation 5-7 in the FCS LTP)

D_C = 25 mrem/yr * (Max $BcSOF_{b,wf}$ + Max $BcSOF_s$ + Max $BcSOF_{b,ep}$ + $BcSOF_{b,f}$ + Max $BcSOF_{bp}$ + $BcSOF_{agb}$ + Max SOF_{egw})

where:

D_C	=	compliance dose (mrem/yr)
Max $BcSOF_{b,wf}$	=	Maximum BcSOF (mean of FSS systematic results plus the dose from any identified elevated areas) for basement wall/floor survey units
Max $BcSOF_s$	=	Maximum BcSOF (mean of FSS systematic results plus the dose from any identified elevated areas) for open land survey units
Max $BcSOF_{b,ep}$	=	Maximum BcSOF (mean of FSS systematic results plus the dose from any identified elevated areas) for basement embedded pipe survey units
$BcSOF_{b,f}$	=	BcSOF (mean of FSS systematic results) for basement fill material open land area survey unit
Max $BcSOF_{bp}$	=	Maximum BcSOF (mean of FSS systematic results plus the dose from any identified elevated areas) from buried piping survey units
Max $BcSOF_{agb}$	=	Maximum BcSOF (mean of FSS systematic results plus the dose from any identified elevated areas) from aboveground building survey units
Max SOF_{egw}	=	Maximum SOF from existing groundwater

6.1.1 <u>Methods for Evaluating Dose and Establishing DCGLs</u>

The methods for evaluating the doses and SOFs from the individual source terms are described in detail in chapter 6 of the LTP. OPPD developed site-specific DCGLs to evaluate the potential dose from residual radioactivity in soil at the site. Similarly, the licensee developed site-specific DCGLs for the basement walls/floors, embedded piping in the basements, buried piping, and basement fill material. OPPD stated that it intends to evaluate the potential dose consequences from groundwater contamination based on the buried piping dose calculations. OPPD chose to use screening values from Table H.1, “Acceptable Screening Levels for Unrestricted Use,” in NUREG-1757, Volume 2, Revision 2, as DCGLs for above-ground buildings.

OPPD derived BcDCGLs for each ROC and for each medium. BcDCGLs are levels of each ROC that would result in a dose equivalent to the dose limit (i.e., 0.25 mSv/yr (25 mrem/yr)) for each ROC and each source. When more than one radionuclide is present, OPPD will apply the sum of fractions rule to ensure that the total dose for a particular survey unit remains within the limit. The sum of fractions methodology takes the radionuclide concentration or activity for each radionuclide present and divides it by the DCGL of the same radionuclide for all the ROCs and sums the ratios. The sum of the ratios of all the ROCs must be less than or equal to one for the combined dose to be less than the dose limit of 0.25 mSv/yr (25 mrem/yr).

The licensee will apply equations 6-1 and 6-2, in the previous section, to demonstrates overall compliance with the unrestricted release criteria of 0.25 mSv/yr (25 mrem/yr) from all sources at the site. The details of how the terms in these equations are determined from sample measurement data are described in subsequent sections of this SER.

In addition, since the BcDCGLs correspond to the level of residual radioactivity that results in a dose of 0.25 mSv/yr (25 mrem/yr) for each medium, the combined dose from all media would be greater than 0.25 mSv/yr (25 mrem/yr) if each medium had residual radioactivity present at these levels. To ensure that the combined dose from all source terms remains under 0.25 mSv/yr (25 mrem/yr), the licensee developed OpDCGLs. OPPD calculated the Operational DCGL values using projected *a priori* dose fractions that it developed based on previous site characterization, process knowledge, and the extent of planned remediation. Table 6-1 lists the *a priori* dose fractions proposed in the LTP. OpDCGLs are calculated by multiplying the BcDCGLs by the *a priori* dose fraction for the corresponding medium.

Table 6-1: *a priori* Dose Fractions used for calculating OpDCGLs Reproduced from LTP Table 5-4

Media	*a priori* Fraction
Basement Floors/Walls	0.15
Basement Embedded Pipe	0.3
Soil	0.28
Buried Pipe	0.14
Above-ground Building	0.09
Basement Fill	0.02
Existing Groundwater	0.02
Sum	**1**

6.1.2 NRC Evaluation of Approach for Overall Dose Compliance

The NRC staff has reviewed the approach and equations in the Fort Calhoun LTP and finds that this approach considers all the source terms (i.e., potentially contaminated media) on site. Additionally, the NRC staff finds that the licensee’s approach of accounting for multiple radionuclides using a sum of fraction approach is consistent with NRC guidance in MARSSIM and NUREG-1757, Volume 2, Revision 2. The NRC staff concludes that the licensee’s plan to use the maximum sum of fractions from any survey unit for all media except for the basement fill material is acceptable because this approach results in a bounding estimate of the dose. The NRC staff concludes that using the average sum of fractions for the basement fill material is acceptable because there is only one survey unit for that material. The NRC staff also finds that assuming that the dose from each medium is additive provides additional conservatism since all of the source terms are not physically located in the same location and therefore a single receptor would not be exposed to all of the sources at the same time. Additional conservatism is also included by assuming that the sum of the dose contributions is based on the value from the highest survey unit for each medium, which are not likely to be located in the same physical area. For these reasons, the NRC staff concludes that the compliance equations in the Fort Calhoun LTP are acceptable for demonstrating that the final combined site dose is less than the unrestricted release criteria of 0.25 mSv/yr (25 mrem/yr).

The NRC staff reviewed the licensee's approach for developing OpDCGLs for each medium. The NRC finds that the licensee's approach of determining base case media specific DCGLs that correspond to a dose of 0.25 mSv/yr (25 mrem/yr) and using projected *a priori* dose fractions to develop OpDCGLs from the BcDCGLs is an acceptable method of ensuring that the final combined dose will be less than 0.25 mSv (25 mrem). The NRC staff notes that the projected *a priori* dose fractions are a key assumption in the development of the OpDCGLs and if there are significant differences in the actual dose fractions, the OpDCGLs might not be adequate to ensure the total combined dose is less than the unrestricted release criteria and additional remediation might be required before completing the final status survey report. Furthermore, the licensee discusses how it will sum the potential dose from each medium in demonstrating compliance with 10 CFR 20.1402 in a logical and reasonable manner. Section 5.9.1 of this SER outlines the process used once FSS data has been collected to ensure that adequate support is provided to support the release of each survey unit.

6.2 Exposure Scenario, Critical Group, and Pathways

Fort Calhoun, located along the Missouri River, is situated in an agricultural region approximately three miles from Blair, Nebraska, and 10 miles from the Omaha, Nebraska metropolitan area. The surrounding area includes an industrial park and recreational areas (e.g., the DeSoto National Wildlife Refuge and Fort Atkinson State Historic Park); no residences are located within one half mile of the facility.

Section 6.3 of the LTP defines the "end state" at the time of license termination as "… the configuration of the remaining below-ground basements, above-ground buildings, piping, and open land areas at the time of license termination." The proposed end state at the time of license termination includes retaining 6 existing aboveground buildings (i.e., Training Building, FLEX Building, Owner Controlled Area (OCA) Entrance Building, 3451 Old Building, 3451 New Building, and the 1251 Control and Switchgear Building), the OPPD Electrical Switchyard and supporting structures, paved roadways and rail lines, and a range of buried pipe, embedded pipe, and penetrations. The basements of the Turbine Building, Containment Building, Auxiliary Building, Intake Structure, and Circulating Water Tunnels will remain, but all interior walls will be removed. The turbine pedestal will remain in the Turbine Building but will be located up to 3 feet below grade. Figure 3.1 of the LTP provides the locations of the basements and aboveground buildings that will remain at the time of license termination. A specific inventory of buried pipe and embedded pipe remaining on site at the time of license termination can be found in OPPD Calculations and Position Paper (CPP) FC-21-002, "Description of Embedded Piping, Penetrations, and Buried Pipe to Remain in Fort Calhoun End State."

6.2.1 <u>Exposure Scenario, Critical Group, and Pathways</u>

Considering current and future conditions at the site, the licensee identified six reasonably foreseeable exposure scenarios. A description of these land-use scenarios along with the environmental pathways and exposure pathways are summarized in table 6-2, which is adapted from table 6-1 and table 6-2 of the LTP.

Table 6-2: Reasonably foreseeable land-use scenarios and associated environmental and exposure pathways

Environmental Pathway	Exposure Pathway	Residential Farmer[1]	Industrial[2]	Residential[3]	Urban Residential / Recreational[4]
Direct Exposure	External Radiation	X	X	X	X
Airborne (Particulate, H-3)	Inhalation	X	X	X	X
Plant Foods	Ingestion	X		X	
Livestock – Meat	Ingestion	X			
Livestock – Milk	Ingestion	X			
Onsite Groundwater	Ingestion	X	X	X	X
Aquatic Foods	Ingestion				
Direct Soil Contact	Ingestion	X	X	X	X

[1]Residential Farmer – Adult that resides onsite and derives a large fraction of annual food intake from onsite agriculture and livestock.
[2]Industrial (includes both Light Industry and Commercial Agriculture) – Adult that works onsite full time, predominantly indoors, performing light industrial activities or outdoors on a commercial farming operation.
[3]Residential – Adult resident that derives a small fraction of annual food intake from an onsite garden. No livestock raised onsite.
[4]Urban Residential/Recreational – The Urban Residential scenario considers an adult resident that does not maintain a garden or livestock whose drinking water is supplied by an offsite municipal source. The Recreational scenario considers an adult that periodically accesses the site (after being converted to parkland or recreation area) for hiking, camping, etc.; No food is harvested from the site, but an onsite well is the drinking water source.

The licensee identified the bounding resident farmer scenario as the average member of the critical group (AMCG) and proposes to use it as the compliance scenario. The scenario assumes that the resident farmer resides in a house constructed on the site and spends the majority of the year onsite conducting subsistence farming activities. Specific transport and exposure pathways are outlined in table 6-2. OPPD also used the bounding industrial worker exposure scenario, which assumes the building may be used for commercial or light industrial activities without any major renovation, to demonstrate compliance for aboveground buildings.

FCS also identified two exposure scenarios that are categorized as less likely but plausible. The first scenario considers a worker installing a water well through the Auxiliary Building or Turbine Building basement. The scenario assumes that pipe cuttings from an embedded pipe are brought to the surface and mixed with the drilling spoils from the borehole. The second scenario considers an offsite worker processing excavated basement concrete and the Containment Building steel liner.

6.2.2 NRC Evaluation of Exposure Scenario, Critical Group, and Pathways

Chapter 5 and appendix I of NUREG-1757, Volume 2, Revision 2, provide guidance on reviewing land-use assumptions. Specific guidance includes considering whether specific critical groups are appropriate for the site and that, within reason, the compliance exposure pathways result in the greatest exposure to the average member of the critical group for all exposure scenarios given the mixture of radionuclides. In addition, the licensee should justify specific future land use(s) based on the nature of the land along with reasonable predictions of future uses. Future uses can be assessed by considering physical and geological characteristics of the site. Licensees should also consider societal uses based on historical information, current uses of the site and similar properties in the area, and what is reasonably foreseeable in the near future.

OPPD provided an adequate basis for using the resident farmer scenario to demonstrate compliance with the unrestricted release criteria of 10 CFR 20.1402, for soil, backfilled basements (including embedded pipe and penetrations), backfill soil, buried pipe, and existing groundwater. OPPD considered that although residential farming is not the most common use of the land immediately surrounding the site, it is common in the area and thus it is reasonable to assume that an individual farmer could settle on the land in the future. In addition, the environmental transport and exposure pathways associated with the resident farmer scenario bound the other reasonably foreseeable exposure scenarios, providing a conservative scenario on which to base compliance. OPPD did not consider the aquatic pathway as an onsite pond does not currently exist and the future construction of one would be hindered by engineering and cost issues. OPPD also noted that an onsite pond is not needed given the site's proximity to the Missouri River. NRC staff find it acceptable to exclude the aquatic pathway and ingestion of aquatic food from an onsite pond given the reasons provided.

In addition to the resident farmer scenario OPPD also considered two industrial worker scenarios for assessing the dose to workers occupying the buildings that will remain on site following license termination. NRC staff also found it acceptable to use the bounding industrial worker exposure scenarios to demonstrate compliance for aboveground buildings since it is consistent with conditions that could be expected on the site following license termination and the assumptions are consistent with the NRC guidance provided in NUREG-1757, Volume 2, Revision 2.

6.3 Source Term

The key areas of review for the source term assumptions are the potential ROCs, configuration, residual radioactivity spatial variability, and chemical form(s) of the source. A review of the potential ROCs, along with potential radionuclides determined to be insignificant contributors to dose, are included in section 5.2 of this SER. The licensee identified seven distinct source terms for the end state of the site which are summarized in the subsections below. Details related to the specific ROCs and specific modeling parameters considered when determining specific source term DCGLs are included in the respective source term DCGL sections below.

6.3.1 Soil

For purposes of the conceptual model the licensee has defined soil as a surface layer to a depth of 0.15 m. The conceptual model also considers soil to a depth of 1.0 m to address the possibility that soil contamination is identified at depths greater than 0.15 m. In both cases the unsaturated zone is still assumed to be 0.1 m below the surface.

The HSA considered documented spills, outage and maintenance activities, and routine operation activities when characterizing soil contamination on the site. The HSA characterizes the majority of the land in the protected area yard, which includes the area surrounding the Containment Building, Auxiliary Building, and support buildings (including the Intake Structure and New Warehouse) as MARSSIM Class 3, indicating that there is a low potential for residual radioactivity to exceed a small fraction of the DCGL. The HSA also notes that spills introducing radioactivity to the soil surface occurred at the railroad siding outside the Auxiliary Building and in the area currently covered by the Radioactive Waste Building. Although records indicate that that remediation occurred and the material was disposed of, the HSA identifies these areas as MARSSIM Class 2 based on the presumption that any remaining radioactivity would not exceed the DCGLs. OPPD identified no plant-derived radionuclides in any of the surface or subsurface samples collected in the open areas in both former protected area and the deconstruction area with the exception of Cs-137, which was present at levels indicative of background levels.

As a result of these findings, the licensee concludes that it is unlikely that significant residual radioactivity is present in the soil in the open land areas or will be present at the time of license termination. The licensee also notes that the soil under the Containment Building, Turbine Building, and Spent Fuel Pool has not been characterized but that any soil contamination in these areas is expected to be localized. Potential soil contamination associated with these areas will be considered as part of continuing characterization activities discussed in section 2.1.3.

6.3.2 Basements

Basements remaining on the site at license termination are associated with the Containment Building, Auxiliary Building, Turbine Building, Intake Structure, and Circulating Water Tunnels. The walls and floors are comprised of concrete and, in the case of the Containment Building, a steel liner. The Auxiliary Building and the Turbine Building also contain embedded pipe. The licensee considered the differences between the various buildings and, based on those differences, calculated one set of DCGL values for the buildings without embedded pipes and a second set of DCGLs for buildings with embedded pipes. The licensee treats both building types separately when calculating the final dose for compliance. Specific details regarding basements associated with the different buildings are included below.

6.3.2.1 Auxiliary Building Basement

The Auxiliary Building basement contains all the support systems for reactor operations that are not located in the Containment Building as well the spent fuel pool, spent fuel support systems, emergency diesel generators, and the emergency core cooling and shutdown cooling systems. As a result, the extent of contamination as well as the radioactive contamination levels vary over the entire area. The licensee noted that this spatial variability—which is described by the gamma survey results provided in the HSA and in table 6-8 of the LTP, should be interpreted cautiously as shine from contaminated equipment and structures present at the time of the survey, but which the licensee plans to remove during decommissioning—may have influenced the data. The licensee also noted that the locations where it collected the concrete wall/floor core samples during characterization were biased to suspect areas, including areas identified as having measurable radioactivity during the survey scans, depressions, discolored areas, cracks, low point gravity drain points, and actual or potential spill locations within the building.

Based on the data collected, the licensee concludes that the general area dose rates averaged over the entire building are lower than the maximum dose rates identified in any given room, indicating that the elevated contamination is likely to be localized and not widespread. The licensee will perform additional analyses to bound the area and depth of the elevated areas. The licensee indicates that it will perform additional investigations to bound the dose rates associated with the elevated areas.

6.3.2.2 Containment Building Basement

The licensee identified significant concentrations of activated concrete under the vessel area during site characterization. However, the licensee plans to remove all concrete down to the steel liner as part of decommissioning activities. As a result, OPPD anticipates the source term at license termination will be limited to widespread surface contamination on the liner at low levels due to cross-contamination from dust generated during the concrete removal process.

6.3.2.3 Turbine Building Basement, Circulating Water Tunnels, and Intake Structure

The licensee concludes that minimal contamination is expected in the Turbine Building basement, Circulating Water Tunnels, and the Intake Structure based on operational history at the site. OPPD detected no plant-derived radionuclides in concrete core samples collected in the Turbine Building basement or the Intake Structure during characterization. OPPD did not collect concrete core samples from the Circulating Water Tunnels due to access issues. Although the liquid waste discharge point is located in the Circulating Water Tunnels, OPPD expects concrete contamination to be minimal due to the low concentrations in the liquid and the high flow rate through the tunnels. The licensee plans on verifying these conclusions during FSS.

6.3.3 <u>Embedded Pipe</u>

Embedded pipe is defined as pipe that runs vertically through a concrete wall or horizontally through a concrete floor. OPPD has identified embedded pipes in the Auxiliary Building and Turbine Building basements and are expected to remain at the time of license termination. From a modeling perspective, the radioactivity on the internal surface of the pipes is assumed to be immediately released from the embedded pipe and mixed with the layer of fill above the floor. This is the same approach considered in the BFM wall/floor model.

OPPD knows the embedded pipes (i.e., drains) in the Auxiliary Building contain elevated levels of radiological contamination and the licensee plans to perform a more detailed assessment survey after performing initial decontamination using high pressure hydrolazing. OPPD expects the Turbine Building embedded pipes to contain minimal contamination and will perform characterization activities as part of the continuing characterization program. However, based on past activities performed in the Turbine Building, OPPD does not expect decontamination of the drains to be required.

The embedded pipes in the Auxiliary Building and the Turbine Building are located at three different elevations. Since the embedded pipe DCGL calculation is dependent on the ratio of embedded pipe internal surface area to total floor area, the licensee calculated separate DCGLs for each floor. The calculated DCGLs for the embedded pipes on different floors is provided in table 6-21 of the LTP.

6.3.4 <u>Buried Pipe</u>

OPPD’s LTP defines buried pipe as pipe located below the ground and outside of structures and basements. OPPD expects the buried pipe, which includes storm drains and service water pipes, to remain on site at license termination. The licensee also plans to keep the service water piping system that serves the maintenance shop. OPPD did not survey the buried pipe during characterization but expects to survey it as part of the continuing characterization program. OPPD does not expect the buried pipe remaining onsite, which was not directly involved with site activities involving radioactivity, to contain residual radioactivity beyond the occasional presence of low concentrations near the radiological detection limits.

6.3.5 Above Ground Buildings

The licensee indicated that the FCS buildings remaining on site at the time of license termination, which are identified as Class 3 structures will be:

- Training Building,
- FLEX Building,
- Owner Controlled Area Entrance Building,
- 3451 Old Building,
- 3451 New Building, and
- 1251 Control and Switchgear Building.

OPPD will demolish all other FCS buildings to a depth of at least 3 feet below grade or removed in their entirety. For the buildings that will remain, surveys performed during characterization did not trigger any scan alarms and results from gross beta static measurements and swab results were at low levels. Due to the low levels of contamination expected, the licensee chose not to establish a reference area for determining net contamination levels.

6.3.6 Groundwater

As noted in EA12-010, "Post 2011 Flood Geotechnical Assessment", groundwater at the site is in hydraulic communication with the adjacent Missouri River, with the water table ranging from 2 to 20 feet below ground surface (bgs) depending on the river stage while “Review of the Groundwater Protection Program at the Fort Calhoun Nuclear Station” (ML21271A151) indicated that, under typical conditions, the depth to groundwater is approximately 15 to 20 feet bgs. Groundwater flow direction at the site has been found to be both toward and away from the Missouri River as a result of seasonal changes and management of river stage levels. However, the groundwater flow rate is very slow due to the low gradients across the site (ML21271A149).

Locally, next to the plant structures, a reverse osmosis (RO) water treatment plant previously withdrew groundwater for use at the plant. This groundwater withdrawal caused a cone of depression and altered groundwater flow beneath the facility buildings. The production well associated with the system was removed from service in 2018. The licensee also performed a search of active groundwater wells in the flood plain on nearby properties to assess potential offsite impacts to the groundwater.

6.3.7 Basement Fill

The LTP identifies the material to be used as basement fill material as the soil that was excavated as part of the rail spur expansion project at FCS. OPPD plans to use the material, which includes approximately 132,000 cubic yards of spoils produced from the excavation of the rail spur area, as fill material after FSS of the structure surfaces and embedded pipe. OPPD excavated the material from a Class 3 impacted area on the FCS site and, thus, it must assess the dose from residual radioactivity.

The LTP notes that the basements will be filled with the soil up to the site grade (1004 feet AMSL). As a result, the material will serve both as fill material within the basement structures as well as the cover soil above the basement structures. Details related to how these uses of the basement fill material impact doses are discussed below.

6.3.8 Chemical Form

The characterization process did not consider the chemical forms of the radiological material in each of the source term materials. To address this, the licensee used the bounding chemical forms (i.e., the chemical form(s) that result in the highest dose per unit intake) as provided in Federal Guidance Report Number 11 (ML101590171), which is consistent with the dose modeling guidance in appendix I of NUREG-1757, Volume 2, Revision 2.

6.3.9 NRC Evaluation of the Source Term Assumptions and Modeling

NRC staff considered the seven distinct source terms discussed in the subsections above and concluded that the licensee provided sufficient detail to ensure an understanding of the possible exposure pathways for consideration when assessing doses associated with the site. As noted above, source term considerations related to ROCs, radionuclides identified as being insignificant contributors, and specific modeling parameters are discussed in other sections of this SER. Additional issues related to the impact of the source term on compliance, such as the use of site-specific soil distribution coefficient (K_d) values and the use of BcDCGLs and OpDCGLs, are also addressed in other areas of the SER.

6.4 Soil Dose Assessment and DCGLs

Section 6.9, "Soil DCGL," of the FCS LTP summarizes the soil conceptual model to be implemented at the FCS site and identifies the exposure scenarios and site-specific soil DCGLs used to evaluate the licensee's compliance with the requirements set forth in 10 CFR Part 50.82(a)(9)(ii)(D), which requires a detailed plans for the final radiation survey to identify the residual radioactivity at license termination. As described in NUREG-1757, Volume 2, Revision 2, the use of target values or DCGLs is a practical method to develop reasonable surveys (i.e., 10 CFR 20.1501(a) and (b)) and to evaluate the dose consequence of survey results (i.e., 10 CFR 20.1402).

6.4.1 Scenarios, Parameters, and Uncertainty Analysis for Soil DCGLs

The surface soil conceptual model, which is associated with the soil dose assessment and derivation of soil DCGLs, used the exposure pathways for the soil source term from the overall site conceptual model, summarized in table 6-2 of the LTP. The surface soil conceptual model assumes a contaminated area of 79,600 m^2 and incorporates all the FCS Class 1 areas, including the entire FCS deconstruction area as well as the waste haul path and loading area. OPPD assumed the waste haul path would be contiguous with the deconstruction area for the purpose of using a contiguous contaminated area. For purposes of the conceptual model, soil contamination is assumed to be uniformly distributed at depths of both 0.15 m and 1.0 m. The 1.0 m thickness is included with the standard 0.15 m thickness to address the possibility that OPPD identifies soil contamination at depths greater than 0.15 m, up to 1 m, during FSS. In both cases, OPPD still assumes the unsaturated zone would be 0.1 m. OPPD does not expect any soil contamination that it may identify to exceed the maximum contamination areas of 1,000 m^2 and 20,000 m^2. OPPD is using the RESRAD family of computer codes to calculate its DCGLs and the computer code has a limit on the maximum contamination areas to calculate doses for the full plant pathways and full meat and milk pathways, respectively. The conceptual model identifies two source release pathways, resuspension to air and leaching to groundwater.

The conceptual model accounts for airborne releases by assuming a constant mass-loading of respirable particles in air with resuspension and deposition in equilibrium. The conceptual model assumes leaching to groundwater would be a function of a constant infiltration rate, constant moisture content, and equilibrium desorption.

OPPD calculated soil dose calculations and corresponding site-specific soil DCGLs for individual radionuclides using RESRAD-ONSITE, version 7.2, and the soil environmental and exposure pathways identified in table 6-2 of the LTP. OPPD used the guidance provided in NUREG/CR-6697 (ML010090252 and ML010090278) and NUREG-1757 to assess the sensitivity and perform uncertainty analyses for the relevant input parameters. OPPD performed the uncertainty analyses separately for each radionuclide. The licensee used this approach to reduce possible influences on the total dose from specific individual radionuclides assumed to be low in quantity. This approach also eliminates potential impacts of uncertainty associated with the mixture fractions. The licensee only performed the uncertainty analyses considering primary contamination at a depth of 1 m but noted that the results are also applicable to the 0.15 m contamination thickness. The results of the uncertainty analysis and the selected deterministic values are summarized in table 6-4, table 6-5, and table 6-6 of the LTP. The licensee noted that when it identified a parameter as being sensitive for one or more radionuclides it assigned the selected deterministic value to all radionuclides, including those that were not determined to be sensitive to the given parameter.

Table A.1.1., "RESRAD Parameters for Soil Uncertainty Analysis," in Attachment 6-1 of the LTP, provides a list of all the RESRAD parameters considered for the uncertainty analysis as well as either the deterministic site-specific values or the range of probabilistic values used. The table also provides the basis for using the selected value and, when applicable, the method used to derive it. During the review NRC staff noted that the licensee used the distribution coefficient (K_d) values for Co-60, Cs-137, and Sr-90 for sand from table 2.13.1 of ANL/EVS/TM-14/4,[1] "Data Collection Handbook to Support Modeling Impacts of Radioactive Material in Soil and Building Structures," when modeling the saturated zone and for loam soil from table 2.13.2 of ANL/EVS/TM-14/4, when modeling the contaminated zone and saturated zone. The licensee used the default RESRAD values included in table 2.13.10 of ANL/EVS/TM-14/4, for C-14 and Eu-152 since those radionuclides are not included in the soil-specific tables. The licensee did not provide the basis for choosing the sand and loam soil types when selecting the DCGLs. In response to RAI TE2-11, which concerns OPPD's use of the reference K_d values to calculate the DCGLs, the licensee notified the NRC that it would be collecting additional soil samples and calculating site-specific K_d values. In the RAI response, OPPD committed to reanalyzing the DCGLs and using the revised values calculated using the site-specific K_d values in place of the published values if they are lower. OPPD stated that changes would only be made to the submitted DCGL values if the site-specific values were determined to be less conservative than the values provided in the LTP but would continue to use the current LTP values if the site-specific values were found to be more conservative.

NRC staff found the general approach for addressing sensitivity and uncertainty to be consistent with NRC guidance provided in NUREG-1757, Volume 2, Revision 2, but considers the use of site-specific K_d values to be more appropriate. Issues associated with the use of site-specific K_d values versus the reference values provided in the LTP are discussed further in section 6.4.3 of this SER.

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[1] Argonne National Laboratory, 2015, "Data Collection Handbook to Support Modeling Impacts of Radioactive Material in Soil and Building Structures," Argonne National Laboratory, Environmental Sciences Division ANL/EVS/TM-12/4

Table 6-7 of the LTP includes the initial suite of soil DCGLs identified by the licensee for depths of 0.15 m and 1.0 m. The final list of BcDCGLs and OpDCGLs from the LTP, adjusted to remove IC dose, are included in table 6-3 and Table 6-4, below. As noted above, these values are based on the reference K_d values for sand and loam provided in ANL/EVS/TM-14/4. NRC staff also noted that OPPD determined Sr-90, which was identified as an ROC for groundwater to be an IC for soils and thus did not include it in the tables. Additional details regarding the exclusion of Sr-90 as an ROC for soils are included in section 5.2.3.

Table 6-3: FCS Soil DCGLs for ROCs (Adjusted for IC Dose)

Radionuclide	0.15 m (pCi/g)	1.0 m (pCi/g)
C-14	5.70E+01	9.68E+00
Co-60	3.77E+00	2.93E+00
Cs-137	1.31E+01	7.27E+00
Eu-152	8.41E+00	7.36E+00

* Multiply picoCuries (pCi) by 3.7×10^{-2} to obtain Becquerels (Bq)

Table 6-4: Operational DCGLs (OpDCGLs) for Soil ROCs (Adjusted for IC Dose)

Radionuclide	0.15 m (pCi/g)	1.0 m (pCi/g)
C-14	1.59E+01	2.71E+00
Co-60	1.06E+00	8.21E-01
Cs-137	3.65E+00	2.04E+00
Eu-152	2.36E+00	2.06E+00

* Multiply picoCuries (pCi) by 3.7×10^{-2} to obtain Becquerels (Bq)

6.4.2 Elevated Areas in Soils and Associated Area Factors

The LTP provides soil area factors as a means of assessing the impact from small areas of elevated radioactivity within the larger cleanup area. As noted in section 5.3 of this SER, the licensee only considered area factors for soil ROCs as it assumes that the surface areas for the excavated concrete would be large and therefore area factors would not be needed.

The licensee calculated area factors for each soil ROC using RESRAD-ONSITE, Version 7.2. OPPD used the same deterministic parameter values used to calculate the soil DCGLs except for the smaller contaminated zone area and corresponding length parallel to aquifer flow parameters. OPPD calculated the area factors by dividing the DCGL values (in pCi/g per 25 mrem/yr) from the smaller areas by the corresponding unadjusted soil DCGLs for each ROC. Table 6-5 includes the soil area factors for each ROC at thicknesses of 0.15 and 1.0 m. OPPD notes that soils in the open land area found with elevated residual radioactivity above the $DCGL_{EMC}$ will be remediated.

NRC staff also noted that the $DCGL_{EMC}$ values for Ni-63 and Sr-90 were calculated by OPPD and included in the supplemental information provided with the LTP, Version 0, but were not listed in the tables of area factors included in the LTP, Revision 0. In response to RAI TE2-14, OPPD informed the NRC staff that area factors, and thus $DCGL_{EMC}$ values, for Ni-63 and Sr-90 were not included in the LTP since they were not identified as ROCs for soils. In addition, the NRC staff also noted that the LTP, Revision 0, included area factors for typical areas considered for these types of reviews as well as an area of 143 m^2. The licensee did not provide the basis for calculating an area factor for this specific area or its use. In response to RAI TE2-14, OPPD informed NRC staff that there is no specific application for an area factor that corresponds to 143 m^2 and that it would remove the values from tables 6-38 and 6-39 of the LTP. OPPD failed to remove the values in Revision 1 of the LTP; however, there is no safety impact associated with its inclusion and thus not of concern to the NRC.

Table 6-5: Soil Area factors for ROCs calculated for 0.15 m and 1.0 m soil thicknesses

	Soil thickness = 0.15 m				
Radionuclide	1 m^2	2 m^2	5 m^2	10 m^2	100 m^2
C-14	3.42E+5	1.43E+05	4.22E+04	1.53E+04	5.89E+02
Co-60	1.23E+01	6.98E+00	3.76E+00	2.48E+00	1.29E+00
Cs-137	1.44E+01	8.25E+00	4.47E+00	2.94E+00	1.56E+00
Eu-152	1.17E+01	6.64E+00	3.59E+00	2.36E+00	1.25E+00
	Soil thickness = 1.0 m				
Radionuclide	1 m^2	2 m^2	5 m^2	10 m^2	100 m^2
C-14	1.69E+05	7.57E+04	2.49E+04	1.03E+04	4.52E+02
Co-60	1.11E+01	6.55E+00	3.61E+00	2.39E+00	1.39E+00
Cs-137	1.95E+01	1.16E+01	6.44E+00	4.28E+00	2.45E+00
Eu-152	9.70E+00	5.74E+00	3.17E+00	2.11E+00	1.23E+00

6.4.3 NRC Evaluation and Independent Analysis of Soil DCGLs and Area Factors

The definition of critical group in 10 CFR 20.1003 is “the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity for an applicable set of circumstances.” The NRC staff concluded that OPPD’s use of the resident farmer as the critical group is reasonable as it reflects the definition and covers all potential exposure routes, thus resulting in more conservative DCGLs than other scenarios. NRC staff also noted that the other reasonably foreseeable land use scenarios considered by OPPD, which are summarized in Table 6-1, “Reasonably Foreseeable Land Use Scenarios, Critical Groups, and Pathways,” of the LTP do not include some of the environmental pathways considered for the resident farmer scenario or have lower occupancy times or food ingestion rates. In addition, the use of the resident farmer scenario is consistent with guidance provided in NUREG-1757, Volume 2, Revision 2, and the NUREG/CR-5512 series. The licensee does exclude the aquatic ingestion pathway associated with an onsite pond from the site-specific resident farmer scenario due to cost and construction issues as a pond would need to be constructed on the site. OPPD also noted the proximity of the site to the Missouri River, which negates any foreseeable need for such a pond. NRC staff found these justifications for not including the aquatic ingestion pathway acceptable. NRC staff also find the licensee’s inclusion of 1.0 m contaminated soil depth, in addition to the 0.15 m contaminated soil depth, appropriate for considering the possibility of additional soil contamination that may be identified during FSS. The NRC staff finds this approach acceptable but notes that the discovery of subsurface soil contamination below the 1.0 m contaminated soil depth or other site characteristics not bounded by the current conceptual mode would trigger the LTP change criteria requirements.

During its review, the NRC staff requested additional information regarding the use of published K_d values for sand and loam soils when evaluating the soil DCGL values provided in the LTP. Specifically, NRC staff questioned the use of separate sand and loam K_d values published in ANL/EVS/TM-14/4, “Data Collection Handbook to Support Modeling Impacts of Radioactive

Material in Soil and Building Structures," for the saturated zone and unsaturated zones (including the contaminated zones), respectively. NRC staff discusses whether the site-specific geological information provided by the licensee was sufficient to support the selection of sand or loam K_d distributions in sections 6.12.3 and 6.12.4 of this SER. As noted above, NRC staff questioned the basis for the use of different soil types in RAI TE2-11. In its response to RAI TE2-11, OPPD noted that it would perform site-specific K_d analyses for each ROC in place of providing a basis for using the referenced K_d values. In the response to RAI TE2-11 OPPD committed to reanalyzing the DCGL calculations using the site-specific K_d values and using the lower DCGLs going forward. OPPD said that itt would continue to use the DCGL values based on reference K_d values included in the LTP if the site-specific DCGLs were found to exceed the reference K_d DCGL values. Section 1.3 describes the process the licensee will use to evaluate whether changes to the DCGL values require a change to the LTP.

As noted in the response to RAI TE2-11, once the reanalysis is complete, OPPD will make a report available to the NRC documenting the approach used and the analyses performed for information and inspection. The report will provide the reanalyzed DCGLs for soil, basement surfaces, embedded pipe, buried pipe, and groundwater using the site-specific K_d values. The report will also document the approach performed by the licensee to determine the K_d values, including the sampling plan used to ensure that the soils collected and analyzed are sufficiently representative of unsaturated and saturated zone soils onsite; the approach used for selecting the analytical, site-specific K_d values to be used in the reanalysis; and the methods used to reanalyze applicable RESRAD files to include the site-specific K_d values, including repeating the probabilistic uncertainty analysis for each ROC after replacing the K_d probabilistic density function with the site-specific K_d values.

NRC staff agrees with the approach of using site-specific K_d values whenever possible, especially when they could be used in place of conservative reference values. However, NRC staff also note that site-specific K_d values that differ from the literature values proposed in the LTP may impact the uncertainty analysis documented in the LTP and result in an inaccurate list of risk-significant parameters. Additionally, since OPPD used the results of the uncertainty analysis to determine parameter values from literature distributions based on the risk-significance of an individual parameter, there is the potential that the basis for using specific values for some parameters may no longer be supported. Thus, the NRC staff agrees with the licensee's approach, as documented in RAI TE2-11, to perform an additional probabilistic uncertainty analysis for each ROC using the site-specific K_d values. This ensures that any changes to the sensitivity of other parameters that could be impacted by the changes to the K_d values are considered.

In addition to reviewing the approach proposed by OPPD, NRC staff performed an independent sensitivity analysis to evaluate the impact from using the range of sand and loam K_d values published in ANL/EVS/TM-14/4 and used by the licensee to calculate the proposed DCGL values in the LTP. NRC staff performed analyses using RESRAD-ONSITE, version 7.2, to compare the DCGL values calculated using the range of 25th and 75th percentile K_d values for sand and loam for each ROC. Results of these independent analyses, summarized in table 6-6 note that changes to the K_d values associated with the different soil types do not appear to impact the DCGLs for individual ROCs for soil. Thus, use of site-specific K_d values is not expected to impact the DCGL values calculated for the site. Table 6-6 compares the DCGL values included in the FCS LTP with the DCGL values calculated using the different percentile sand and loam soil K_d values. Although the use of site-specific Kd values is preferred whenever possible, based on the results summarized in table 6-6 the NRC staff does not expect OPPD's use of different reference sand and loam soil K_d values to significantly impact the overall dose.

For this reason the NRC staff concludes that the DCGL values proposed by the licensee for soil are acceptable for demonstrating compliance with the unrestricted release criteria in 10 CFR 20.1402.

Table 6-6: Comparison of Base Case DCGLs (BcDCGLs) for soil using a range of K_d values for contaminant depths of 0.15 m and 1.0 m (values not adjusted for IC dose)

0.15 m Base Case Soil DCGL (Not adjusted for IC Dose)[1] (pCi/g)				
	C-14	Co-60	Cs-137	Eu-152
Reported DCGL (from LTP)	6.00E+01	3.97E+00	1.37E+01	8.86E+00
75th percentile Loam CZ/UZ[2] and 75th percentile Sand SZ	6.00E+01	3.97E+00	1.37E+01	8.86E+00
25th percentile Loam CZ/UZ and 25th percentile Sand SZ	6.00E+01	3.97E+00	1.37E+01	8.86E+00
75th percentile Loam for CZ/UZ and SZ	6.00E+01	3.97E+00	1.37E+01	8.86E+00
25th percentile Loam for CZ/UZ and SZ	6.00E+01	3.97E+00	1.37E+01	8.86E+00
75th percentile Sand for CZ/UZ and SZ	6.00E+01	3.97E+00	1.37E+01	8.86E+00
25th percentile Sand for CZ/UZ and SZ	6.00E+01	3.98E+00	1.38E+01	8.86E+00
1.0 m Base Case Soil DCGL (Not adjusted for IC Dose) (pCi/g)				
	C-14	Co-60	Cs-137	Eu-152
Reported DCGL (from LTP)	1.02E+01	3.09E+00	7.66E+00	7.75E+00
75th percentile Loam CZ/UZ and 75th percentile Sand SZ	1.02E+01	3.09E+00	7.66E+00	7.75E+00
25th percentile Loam CZ/UZ and 25th percentile Sand SZ	1.02E+01	3.09E+00	7.66E+00	7.75E+00
75th percentile Loam for CZ/UZ and SZ	1.02E+01	3.09E+00	7.66E+00	7.75E+00
25th percentile Loam for CZ/UZ and SZ	1.02E+01	3.09E+00	7.66E+00	7.75E+00
75th percentile Sand for CZ/UZ and SZ	1.02E+01	3.09E+00	7.66E+00	7.75E+00
25th percentile Sand for CZ/UZ and SZ	1.02E+01	3.09E+00	7.66E+00	7.75E+00

* Multiply picoCuries (pCi) by 3.7×10^{-2} to obtain Becquerels (Bq)

[1] Adjusted DCGL values can be calculated by multiplying values by 0.95

[2] CZ = Contaminated zone; UZ = Unsaturated zone; SZ = Saturated zone

In addition to the K_d issues, NRC staff also reviewed the proposed soil area factor values discussed in section 6.4.2 and listed in table 6-5 of this SER. NRC staff found the approach used to derive the values to be acceptable and, as noted in section 6.4.2, found no safety issues associated with the licensee's exclusion of area factor values for an area of 143 m^2 in Revision 1 of the LTP. Additional details regarding the NRC staff's review of the application of these values during FSS are discussed in section 5.5.2 of this SER.

6.5 Basement Walls and Floor Dose Assessment and DCGLs

Sections 6.10, "Backfilled Basement Model Overview", 6.11, "BFM Wall/floor Scenario DCGL," and 6.12, "BFM Wall/Floor Initial Suite DCGL" in the FCS LTP describe the conceptual model and calculations used to derive DCGLs for the backfilled basements that will remain at the FCS site at the time of license termination. According to the LTP, the basements to remain are the basements for the Containment Building, Auxiliary Building, Turbine Building, Intake Structure, and the Circulating Water Tunnels. The licensee states that it will remove the buildings to a minimum of 3 ft below grade (approximately 1,001 feet AMSL). The licensee commits to removing all interior walls of these basements, with the exception of the turbine pedestal in the Turbine Building, which will remain up to a height of 3 ft below grade. The basements will be backfilled to grade level (1,004 feet AMSL) once D&D and FSS activities are completed. Per the LTP, all basement walls/floors will therefore have a minimum clean cover thickness of 3 ft (0.92 m). OPPD assumes the source term in the basements would be a volumetric or surface layer with a uniform distribution.

The basements include contaminated walls and floors as a medium or source term. The walls and floors are comprised of concrete. The containment building also has a steel liner. OPPD generated one set of DCGLs for the walls and floors in the Auxiliary Building, Turbine Building, Intake Structure, and Circulating Water Tunnel Basements. OPPD calculated a separate DCGL for the walls and floors in the Containment building due to the presence of the steel liner in it. OPPD used the wall/floor DCGL values for penetrations within the basements.

According to the LTP, the BFM wall/floor model includes three source release pathways:

- Instant release from the concrete or steel containment liner to the water in the fill material pore space (referred to as the *in situ* scenario),
- Capture of concrete or steel liner in drilling spoils generated during the installation of an onsite well (drilling spoils scenario), and
- Excavation of concrete walls or steel liner (excavation scenario).

The licensee stated that it is summing the doses from these three release pathways to calculate the final wall/floor DCGL using the following equation.

Equation 6-3: BFM Wall Floor DCGL Calculation (Equation 6-13 in the FCS LTP)

$$DCGL_{wf} = \frac{1}{\left(\frac{1}{DCGL_i} + \frac{1}{DCGL_{ds}} + \frac{1}{DCGL_e}\right)}$$

where:

$DCGL_{wf}$	=	BFM wall/floor DCGL
$DCGL_i$	=	BFM *in situ* scenario DCGL
$DCGL_{ds}$	=	BFM drilling spoils scenario DCGL
$DCGL_e$	=	BFM excavation scenario DCGL

According to the LTP, the Auxiliary and Turbine Building basements also contain embedded pipes. The licensee stated that because these embedded pipes are significantly different from the walls/floors in terms of their source terms, physical configurations, conceptual models, and source term abstraction and release, OPPD treated the embedded pipes as separate media and are discussed in section 6.9 of this SER.

6.5.1 *In situ* Scenario Dose Modeling

In the *in situ* BFM scenario, the receptor uses well water that was contaminated as a result of leaching of the residual radioactivity from the backfilled concrete surfaces and steel liner in the containment building. OPPD assumed the well water would be used to support a farm. OPPD also assumed exposure pathways would be direct exposure, inhalation, and ingestion of plants, meat, milk, soil, and drinking water. In this scenario, the release of this source term then leads to contamination of the fill material and the water in the pore space of the fill material. OPPD assumed all of the contamination on the walls and floors would be released instantly and mix uniformly between the wall and the groundwater well. OPPD assumed the groundwater well would be located at a distance of 1 m (3.3 ft) from the wall. The license stated that the instantaneous release assumption was a conservative assumption because the residual radioactivity would release more slowly by diffusion. The licensee indicated that the slower release from diffusion would allow for radioactive decay and source depletion as the water containing radioactivity is removed through the groundwater well. NRC staff agrees that instantaneous release is conservative relative to the realistic physical release mechanism of diffusion because (i) any delay in release leads to lower doses, and (ii) total dose estimates are highest at license termination and decrease over the 1,000-year performance period.

In the following subsections, staff (i) describes the approach taken for the development of BFM *in situ* DCGLs, (ii) describes and reviews the RESRAD-ONSITE representation, (iii) describes and reviews the Groundwater Vistas model, (iv) compares the results of the RESRAD-ONSITE representation and Groundwater Vistas models, and (v) provides a conclusion for the adequacy of the BFM *in situ* DCGL values.

6.5.1.1 Development of BFM in situ DCGLs

Section 6.11 of the LTP provides details of the calculation of DCGLs for the *in situ* scenario. The licensee used RESRAD-ONSITE to calculate dose-to-source ratios (DSRs) for the *in situ* scenario, which it then used to calculate DCGLs for this scenario that correspond to a dose of 0.25 mSv/yr (25 mrem/yr). The licensee implemented RESRAD-ONSITE using the option to submerge the entire rectilinear prism-shaped contaminated zone in the saturated zone. The RESRAD-ONSITE abstraction assumes horizontal flow in the saturated zone where flow is unimpeded through the contaminated zone and across the subsurface structure footprint. Because RESRAD-ONSITE does not adequately represent the backfilled subsurface structure of the *in situ* BFM scenario, the licensee chose to perform offline modeling to support DCGL estimates. In section 6.11 of the LTP, the licensee provided descriptions and results of offline numerical modeling of flow and transport used to confirm that the well water concentration results from the RESRAD-ONSITE BFM *in situ* model abstraction were not underestimated. In section 6.11.3 of the LTP, the licensee briefly summarized Groundwater Vistas simulations and pointed to descriptions provided in appendix F of the Haley & Aldrich (2021) report titled, “Hydrogeological Conceptual Site Model, Revision 5 Fort Calhoun Station Blair Nebraska.” Haley & Aldrich (2021) provided detailed descriptions of the flow and transport model input and results of sensitivity analyses obtained using the Groundwater Vistas interface for the MODFLOW groundwater flow code and MT3DMS transport code. Additionally, based on analyses reported in Haley & Aldrich (2021), the licensee included in LTP Attachment 6.2 illustrations of streamlines (pathways) of radionuclides derived from simulations using the MODPATH particle tracking code.

NRC staff notes that there are three aspects of backfilled subsurface structures that cannot be reasonably represented in RESRAD-ONSITE: (i) the release of residual radioactivity in cement into backfill where a pumping well is located, (ii) the configuration of the contaminated zone along walls and a floor, and (iii) three-dimensional flow and transport in a substructure filled with backfill. For the first aspect, the licensee conservatively assumed that the radionuclide inventory in the concrete of walls and floors instantaneously releases and mixes in a 1-m (3.3-ft) wide zone of backfill adjacent to those walls and floors. Staff notes that this is a conservative assumption because diffusion of radionuclides from the concrete to the adjacent backfill would greatly reduce peak radionuclide concentrations in the backfill and subsequently become available to a pumping well. For the second and third aspects, the licensee used the offline Groundwater Vistas model to more realistically represent the flow and transport in the backfilled subsurface structure. For the second aspect, the contaminated zone in the RESRAD-ONSITE abstraction is limited to a horizontal rectangular contaminated source area. In Groundwater Vistas, the contaminated zone can represent the contaminated zone as a three-dimensional shape such as a 1-m (3.3-ft) thick zone adjacent to both walls and a floor. For the third aspect, RESRAD-ONSITE cannot explicitly incorporate the effect of the three-dimensional geometry of the walls and floor on flow and transport. Groundwater Vistas can simulate (i) a bathtub model where floor and walls do not allow exchange of water through the concrete, (ii) walls and floors that are permeable to flow due to cracks, joints, or total degradation of the concrete, thus the aquifer flows may flow unimpeded across the structure, and (iii) the placement of wells anywhere within or outside the subsurface structure.

6.5.1.2 RESRAD BFM In Situ Model

The licensee modeled the backfilled basement structures in RESRAD-ONSITE using a contaminated zone submerged in the saturated zone with the nondispersion transport model enabled. Depending on the recharge (vertical) and horizontal flux rates and the pumping rate, the nondispersion transport model allows for some dilution of radionuclides in the well water compared to the Mass Balance transport model.

In table 6-10 of the LTP, the licensee provided a short list of changes from values used for the soil DCGL to values applied in the BFM *in situ* DCGL. For the BFM *in situ* model, the licensee changed RESRAD inputs linked to the contaminated zone and the pumping well. For the source term geometry, the licensee reduced the contaminated zone area to 16,700 m^2 (179,800 ft^2) and the length parallel to flow to 70 m (750 ft) compared to the values used for the soil DCGL model. According to section 6.11.3 of the LTP, the licensee based the 16,700 m^2 (179,800 ft^2) area on the total basement surface area for the subsurface structures and the 4 m thickness on the end state height of the basement walls for the Auxiliary and Turbine buildings. Staff note that the contaminated zone dimensions in table 6-9 of the LTP, confirmed by staff in the licensee-supplied RESRAD-ONSITE input file for the BFM *in situ*, lead to a 239-m (784-ft) wide contaminated zone. But the licensee stated in section 6.11.3 of the LTP that it assumed the contaminated source zone would be 60 m (197 ft) wide, consistent with the widths of both the Auxiliary and Turbine buildings. For the cover overlying the contaminated zone, the licensee set the erosion rate to zero and reduced the thickness to 0.92 m (3 ft) to account for the assumption that the entire subsurface structure, backfill, and contaminated zone were completely submerged in the saturated zone. For hydrologic parameters of the contaminated zone, the licensee changed from unsaturated zone values used in the soil DCGL model to values consistent with the saturated zone values in the soil DCGL model.

The BFM *in situ* model used sorption coefficients (K_d) both for the contaminant source release model and for transport to the pumping well. OPPD based the K_d value for the contaminated, unsaturated, and saturated zones on distributions from the sand and generic (C-14 only) tables in the Data Collection Handbook (ANL, 2015, table 2.13.1 and 2.13.10, respectively). Based on uncertainty analyses using RESRAD-ONSITE, the licensee indicated that dose was sensitive and negatively correlated to sorption coefficient values, and thus, set the Kds to the 25th percentile values as shown in table 6-12 of the LTP for subsequent DCGL calculations.

Using the input file for the BFM *in situ* scenario release pathway provide by the licensee, staff noted the RESRAD-ONSITE output for a dilution factor was 0.766 for the well pumping rate of 4,550 cubic meters per year (m^3/yr) (160,700 cubic feet per year (ft^3/yr)) and configuration of the contaminated zone. Other factors affecting calculations of dilution include groundwater flow rates and well depth, for which the licensee used site-specific values. The dilution factor is the ratio of the groundwater concentration in the contaminated zone to the concentration in the pumping well. Staff notes that reducing dilution, which corresponds to larger dilution factor in RESRAD-ONSITE, is more conservative, i.e., a dilution factor equal to 1 indicates no dilution and is the maximum value possible in RESRAD-ONSITE. Staff will compare this dilution factor of 0.766 from RESRAD-ONSITE to results from the Groundwater Vistas model in the next section to ensure that the RESRAD BFM *in situ* model does not underestimate well water concentration.

As part of its review and probing of the licensee's approach, staff reproduced the DCGL values provided in table 6-13 of the LTP to confirm that staff was using the correct RESRAD-ONSITE input file. Whereas staff confirmed that the RESRAD-ONSITE output values for dose to source ratios led to calculated DCGL values consistent with the licensee's DCGL values in table 6-13 of the LTP, staff noticed that RESRAD-ONSITE code did not correctly incorporate the dilution factor into the well concentrations prior to calculating dose. This error in the code occurs when the combination of a submerged contaminated zone and the non-dispersion transport model are selected in RESRAD-ONSITE. The consequence of this RESRAD-ONSITE code error is that OPPD may have underestimated the DCGL for the BFM *in situ* pathway due to the dilution factor not being incorporated into the well water concentration. Because the RESRAD-ONSITE error leads to smaller DCGL values, staff notes that the BFM *in situ* DCGL values in table 6-13 of the LTP are conservative. Because the licensee's DCGL values are conservative, staff finds the RESRAD-ONSITE error does not affect the use of the DCGL values in the FSS.

6.5.1.3 The Groundwater Vistas Model

The licensee provided the description of the flow and transport BFM, herein referred to as the Groundwater Vistas Model, and results in appendix F of Haley & Aldrich (2021, ML22034A594). Haley & Aldich (2021) indicated that the BFM was implemented using MODFLOW-2005

(Harbaugh, 2005)[2] and MT3DMS (Zheng and Wang, 1999)[3] for flow and transport, respectively. Staff notes that input files provided by the licensee used MODFLOW-6 (Langevin et al., 2017)[4], which is an integrated finite difference formulation rather than the classical finite difference formulation of MODFLOW-2005. Staff expects that results of the flow component would not significantly change using either the MODFLOW-2005 or MODFLOW-6 flow code for the BFM as implemented. This expectation is based on staff's expert judgment considering grid domain, cell sizes, boundary conditions, hydrological input values, and the nuances of the different formulations.

In Haley & Aldrich (2021), the licensee described the implementation of flow and transport BFM as a three-dimensional finite difference grid covering a horizontal area of 130 m by 130 m (427 ft by 427 ft) and a vertical length of 30 m (98 ft) divided into a total of 33,000 cells in a non-uniform, rectilinear grid. The licensee defined the base model dimensions of the subsurface concrete structure as a 60 m by 60 m (197 ft by 197 ft) footprint with walls that extend to the 3 feet (0.92 m) below the ground surface, either as the base case 9-m (30-ft) wall or an alternative 4-m (13-ft) wall. The licensee used the properties for alluvial sediments for the upper 20 m (67 ft) of the domain and properties for limestone bedrock for the lower 10 m (33 ft) of the domain.

For the flow portion of the Groundwater Vistas model, the licensee reported that it set hydraulic conductivity, porosity, head gradient inputs consistent with the parameter values in the RESRAD BFM *in situ* model. Based on the specified boundary conditions, staff deduced that flow was steady state and transport was transient. The boundary conditions included specified head on the west and east boundaries to impose a head gradient of 0.00084 across the domain, specified recharge across the model domain, and a single pumping well with a constant rate of 4,550 m^3/yr (160,700 ft^3/yr). According to appendix F of Haley & Aldrich (2021), the licensee set recharge across the model to 0.076 m/yr (3 inches/yr), except for over the contaminated zone within the subsurface structure where the recharge was set equal to 1.2 m/yr (48.7 inches/yr). Staff notes that setting the recharge rate over the structure equal to the pumping rate ensured that the backfill in the substructure would not dewater in the bathtub scheme when the concrete was treated as no-flow condition.

In the Groundwater Vistas transport model, the licensee reported on results for Cs-137 and Sr-90 as examples with different K_d values. According to appendix F of Haley & Aldrich (2021), the licensee set the dispersivity parameter value to zero, enabled first-order degradation rate constant to account for radioactive decay, and set K_d values for cesium and strontium to 158 and 6.6 liters per kilogram (L/kg), respectively. The licensee implemented upstream weighting, which Haley & Aldrich (2021) indicated creates some amount of numerical dispersion. The licensee implemented the contaminated zone as an initial condition set to a unit concentration in the pore water of the 1-m (3.3-ft) wide backfill zone adjacent to walls and floors. As an initial condition, recharge and pumping would flush the contaminated zone. Staff notes that the resulting well water concentrations scale to the pore water concentrations in the contaminated zone.

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[2] Harbaugh, A.W., 2005, "MODFLOW-2005, the U.S. Geological Survey modular ground-water model -- the Ground-Water Flow Process," U.S. Geological Survey Techniques and Methods 6-A16.
[3] Zheng, C. and P.P. Wang, 1999, "MT3DMS: A Modular Three-Dimensional Multispecies Transport Model for Simulation of Advection, Dispersion, and Chemical Reactions of Contaminants in Groundwater Systems; Documentation and User's Guide," Contract Report SERDP-99-1, U.S. Army Engineer Research and Development Center, Vicksburg, MS.
[4] Langevin, C.D., Hughes, J.D., Banta, E.R., Niswonger, R.G., Panday, Sorab, and Provost, A.M., 2017, "Documentation for the MODFLOW 6 Groundwater Flow Model," U.S. Geological Survey Techniques and Methods, book 6, chap. A55, 197 p., https://doi.org/10.3133/tm6A55.

For the contaminated zone geometry, RESRAD-ONSITE is limited to a rectangular prism and flow through the contaminated zone is one-dimensional. The Groundwater Vistas model can represent the 1-m volume adjacent to both walls and floors and the interplay of that three-dimensionally complex contaminated zone with a pumping well drawing water from different locations within the subsurface structure. In addition, Groundwater Vistas can treat the concrete walls and floor as either intact using a no-flow condition or as degraded by setting the permeability equal to the surrounding sediments such that groundwater flow is not impeded.

In section 6.11.2 of the LTP, the licensee described the ratio of concrete surface area to backfill volume ratio (SA/V) for each subsurface structure. The license stated that the SA/V equal to one is the full mix assumption, which staff takes to mean the entire volume of 1-m wide zone of contaminated backfill incorporates the entire structure volume in an average sense whereby the volume of the 1-m wide zones of backfill that overlap between adjacent floors and walls is offset by volumes not occupied by any of the 1-m wide zones. The licensee reported SA/V for Auxiliary Building basement, Containment Building basement, Turbine Building basement, Intake Structure, and Circulating Water Tunnels of 0.37, 0.65, 0.50, 0.41, and 1.05, respectively. Noting that the SA/V value for the Circulating Water Tunnels was greater than the full mix assumption, the licensee stated in section 6.11.3 of the LTP that the exceedance of 5 percent was trivial compared to the conservatism in the BFM and the expectation of minimal contamination in the Circulating Water Tunnel. Staff concurs that the exceedance is trivial both for the reasons stated by the licensee and because the Circulating Water Tunnels encompass a much smaller area compared to the any of the other backfilled subsurface structures. For the Groundwater Vistas model, Haley & Aldrich (2021) implemented two footprint sizes and two wall heights; 60 m by 60 m (197 ft by 197 ft) and 30 m by 30 m (98 ft by 98 ft) footprints, and 9-m and 4-m (30-ft and 13-ft) walls. The wall heights cover the range exhibited by the larger heights of walls for the Containment Building compared to the smaller heights of wall for the Auxiliary or Turbine Buildings. Staff calculated the SA/V ratios for the Groundwater Vistas permutations as ranging from 0.18 to 0.38. Whereas the Groundwater Vistas SA/V values do not provide a bounding confirmation, staff notes that results are likely valid for all buildings with SA/V values significantly below a value of one because the concentration results provided in table 6-9 of the LTP do not change significantly when the building footprint or wall height were reduced. Therefore, staff finds that the Groundwater Vistas model representation of backfilled subsurface structures will lead to results approximating or bounding realistic conditions at the site because most of the remaining structures have SA/V values significantly below one and sensitivity is small for changes to wall heights and area of building footprint.

The licensee provided results of a sensitivity analysis for all permutations of the Groundwater Vistas models in table 6-9 of the LTP to address uncertainty and variability. Haley & Aldrich (2021) described a range of configurations that included permeability of concrete, well placement location, well pumping rate, well depth, subsurface structure wall height and floor footprint, and source thickness. The dilution factors for the Groundwater Vistas cases, (which are equal to the ratio of model results from MT3D and RESRAD BFM *in situ* in table 6-9 of the LTP) ranged from 0.006 to 0.70. Staff notes that the largest reported value, 0.70, pertained to the 60 m by 60 m footprint with 4-m walls with the concrete walls not impeding the natural groundwater flow. The bathtub case where concrete impedes groundwater flow (the no-flow condition) resulted in slightly more dilution (smaller dilution factor) compared to the unimpeded groundwater condition. Comparison of the Groundwater Vistas model results with those of the RESRAD-ONSITE BFM *in situ* results are discussed in the next subsection.

6.5.1.4 Comparison of MT3D and RESRAD Results

In table 6-9 of the LTP, the licensee provided ratios of MT3D to RESRAD-ONSITE well water concentration results for a variety of scenarios implemented in the Groundwater Vistas model. Staff noted that well water concentration results reported in Haley & Aldrich (2021) for the Groundwater Vistas model were based on using a contaminant source area with unit radionuclide concentrations in the pore water, i.e., contaminant concentrations in 1-m zone adjacent to walls were set to an initial condition equal to 1 mg/L. Because well water concentrations scale to source area concentrations, the well water concentration results from the Groundwater Vistas model are also equal to the MT3D to RESRAD-ONSITE BFM *in situ* model ratios of concentration results in table 6-9 of the LTP, which is equal to the dilution factor. A dilution factor from the Groundwater Vistas simulations that was greater than the RESRAD *in situ* result of 0.766 would indicate that RESRAD-ONSITE BFM *in situ* model was underestimating well water concentrations.

The licensee stated that all of the Groundwater Vistas configurations and conditions resulted in dilution factors less than 0.766. For the most direct comparison, staff compared the configuration where concrete walls and floors do not impede groundwater flow in Groundwater Vistas, i.e., concrete permeability is set to that of the natural sediments and backfill. This configuration matches the abstraction in RESRAD-ONSITE where the groundwater flow is unimpeded. Based on table 6-9 of the LTP, the Groundwater Vistas dilution factor is 0.45 when the well is placed along the center of a wall and 0.65 when placed in the corner. For the bathtub condition implementation, where walls and floor were approximated as impermeable, results in table 6-9 of the LTP indicated a dilution factor of 0.61 for a well placed in the corner; or, slightly more dilution compared to the case where walls and floor do not impede groundwater floor. The base case in the Groundwater Vistas model is implemented with 9-m walls. For 4-m wall height in table 6-9 of the LTP with the well placed in the corner, the licensee reported a dilution factor of 0.70 for unimpeded groundwater flow and 0.65 for the bathtub configuration. Staff notes that the corner placement of wells is conservative compared to locating them somewhere else along the wall, or anywhere closer to the middle of the subsurface structure.

In section 6.10.2 of the LTP, the licensee suggested that any uncertainty in the comparison of results from the RESRAD-ONSITE and Groundwater Vistas models is not important due to the small dose amount that BFM *in situ* model contributes to the 0.25 mSv/yr (25 mrem/yr) criteria. The BFM includes separate DCGLs for two media: walls/floors (section 6.11 of the LTP) and embedded pipes (section 6.13 of the LTP). The BFM wall/floor media has three separate source release pathways that require separate DCGLs: *in situ*, excavation, and drilling spoils. The DCGL for wall/floor is a summation of DCGLs calculated for the *in situ*, excavation, and drilling spoils based on LTP Equation 6-13. The licensee provided an analysis in section 6.11.2 of the LTP using Cs-137 that showed the BFM *in situ* DCGL contributed 29 percent to the DCGL for walls and floors. In section 6.11.2 of the LTP, the licensee combined this percent with the allotted portion of 0.3 for the DCGL for walls and floors to indicate a dose of 0.022 mSv/yr (2.2 mrem/yr) associated with the BFM *in situ* DCGL. The licensee suggested that this small dose for the BFM *in situ* DCGL relative to the 0.25 mSv/yr (25 mrem/yr) release criteria reduced the dose impact of uncertainty in the BFM *in situ* model results. Staff notes that table 5-4 of the LTP allotted a portion of 0.15 to BFM wall/floor DCGL, not 0.3. Therefore, the dose associated with the BFM *in situ* DCGL would be smaller, and thus the uncertainty less important than the example the licensee provided in section 6.11.2 of the LTP. Additionally, the licensee suggested in section 6.11.2 of the LTP that uncertainty was not important because of the conservative assumptions made in the BFM *in situ* model. The licensee pointed to two conservative assumptions: (i) the instantaneous transfer of radionuclides from the concrete to the 1-m (3.3-ft) wide zone of backfill adjacent to walls and floors is conservative compared to the diffusion process out of the concrete and dispersion into the backfill, and (ii), the likelihood of the placement of a well in or adjacent to the 1-m (3.3-ft) zone near substructure walls is less that 3 percent based on the ratio of contaminated area and site area.

6.5.1.5 NRC Evaluation BFM In Situ DCGLs

The NRC staff reviewed the calculation of the BFM *in situ* DCGLs and concludes that the scenario and receptor assumptions are appropriate for evaluating the potential dose from residual radioactivity released from the basement walls and floors to water in the basements because these assumptions are consistent with the site configuration and potential future uses. The NRC staff further evaluated whether the simplified groundwater modeling abstraction in RESRAD was appropriate for evaluating the dose from residual radioactivity released to the groundwater. The licensee stated in section 6.11.3 of the LTP and provided summary results in table 6-9 of the LTP showing that results from RESRAD BFM *in situ* model did not underestimate well water concentrations of radionuclides based on a comparison with results from the Groundwater Vistas model. The Groundwater Vistas model simulated a variety of configurations and conditions, including permeability of concrete walls and floor, substructure size, well placement location, and pumping rate. Therefore, NRC staff finds that the well water radionuclide concentrations are acceptable and are not underestimated by the RESRAD BFM *in situ* model based on staff's analysis of the licensee's model development and results from offline detailed numerical flow and transport codes. The NRC staff also performed independent calculations of the BFM *in situ* DCGLs in RESRAD and obtained comparable results. For these reasons, the NRC staff concludes that the BFM *in situ* DCGLs provided in the FCS LTP are acceptable for evaluating the potential dose contribution to the Basement Walls and Floor DCGLs.

6.5.2 <u>Drilling Spoils Scenario Dose Modeling</u>

Sections 6.11.7, 6.11.8, and 6.11.9 of the LTP describe the methodology OPPD used to calculate the Drilling Spoils scenario DCGLs. In the Drilling Spoils scenario, the drilling spoils from an 8-inch diameter borehole and are assumed to be spread on the ground surface in a 0.15 m thick layer. The licensee assumed that the drill would stop one inch after meeting refusal from the concrete floor. The licensee stated that the minimum distance between the ground surface and a floor in the basements for the Auxiliary Building, turbine Building, or Containment Building is 8.5 feet (2.59 m), which corresponds to a drilling depth of 2.62 m (8.5 feet + 1 inch). This minimum distance was used to calculate the volume of the drilling spoils that OPPD modeled as being brought to the surface.

The drilling spoils that OPPD has brought to the surface are modeled as surface soil in RESRAD. The licensee stated that it used the same deterministic parameter values as in the development of the soil DCGLs, with the exception of the area of the contaminated zone, thickness of the contaminated zone, and the length parallel to the aquifer flow to develop DSRs for the drilling spoils. OPPD calculated the area of the contaminated zone by dividing the volume of spoils by the assumed 0.15 m thickness. OPPD assumed the length parallel to the aquifer flow would be equal to the diameter of the resulting circular area. As in the case of the *in situ* scenario, OPPD assumed the residual radioactivity on 1 m^2 of wall surface would be mixed into 1 m^3 of fill. OPPD used the DSRs and the mixing assumptions to develop BcDCGLs for the Building Spoils scenario. The licensee provided these DCGLs in table 6-15 of the LTP.

6.5.2.1 NRC Evaluation of Drilling Spoils DCGLs

The NRC staff reviewed the methodology and parameter values used in the development of the Drill Spoils scenario DCGLs. The NRC staff finds that the assumption that the spoils will have the same properties as soil is appropriate given that most of the backfill is soil and most of the material that would be brought up as drilling spoils would be soil. The modeling of any concrete that is brought up to the surface as soil is conservative since it is less likely that farming would occur in these cementitious materials. If the future land use did not include farming, many of the exposure pathways would not exist (e.g., ingestion of plants, meat, milk) and therefore the overall potential dose would be less. The NRC staff finds that the use of the RESRAD code and the same parameter values as used in the development of the soil DCGLs is appropriate given that both scenarios apply to the same site and the parameters would be expected to be the same. The NRC staff finds that the assumptions regarding the bore hole thickness, depth of drilling, volume of drilling spoils, and thickness of the spoils once placed on the ground are appropriate because they are consistent with the assumed scenario and the site conditions. The NRC staff reviewed the calculation of the DSRs and the calculation of the DCGLs from the DSRs and obtained comparable results. For the reasons described in this paragraph, the NRC staff concludes that the use of the Drill Spoils DCGLs is acceptable for calculating the contribution of the Drill Spoils scenario to the Basement Walls and Floor DCGLs.

6.5.3 Excavation Scenario Dose Modeling

Section 6.11.10 of the LTP provides the details of the development of the excavation scenario DCGLs. In this scenario, OPPD assumed that material from a basement would be excavated and brought to the surface. For all basements except for the Containment Building, the material that OPPD assumed would be brought to the surface is concrete wall. For the Containment Building, OPPD assumed that the steel liner would be excavated. For the excavation of a concrete wall, OPPD assumed that a wall with the minimum thickness on site (i.e., 2 ft or 0.61 m) would be excavated and spread over a 1 m depth on the ground surface. OPPD also assumed that in the process of excavating the concrete, the material would be sized to allow for use as onsite backfill, that the contaminated and uncontaminated portions of the concrete would be mixed, and that any rebar in the concrete would be removed. OPPD then modeled the material as having the same characteristics as soil. For the steel liner, OPPD assumed that mixing within the liner would not occur during excavation. Instead, it assumed any residual radioactivity on the liner would be instantaneously released to the soil in an area equal to the area of the liner that it excavated. In the LTP, the licensee performed a sensitivity analysis with both a soil mixing depth of 0.15 m and 1 m for the residual radioactivity released from the liner. The licensee found that assuming a mixing depth of 0.15 m resulted in the minimum value for the DCGLs, so it assumed the mixing depth would be 0.15 m. In the LTP, the licensee also performed a check to confirm the assumption that the excavation of the concrete wall is conservative compared to assuming that it excavated fill material. The licensee confirmed that the modeled concentration in the fill is less than the concentration in the concrete and, therefore, concluded that assuming the source term in the excavation scenario is the concrete wall rather than the fill is conservative.

The licensee used the DCFs that were generated for the resident farmer scenario for soil (section 6.4 of this SER) to evaluate the dose from the residual radioactivity from the excavated material. OPPD used these DCFs and the assumptions regarding the geometry of the excavated material and mixing depth to develop excavation DCGLs. OPPD used Equations 6-8 and 6-9 in the LTP for the concrete material and Equations 6-10 and 6-11 in the LTP for the steel liner. The resulting BFM excavation DCGLs for the whole suite of radionuclides without an

IC correction are in tables 6-16 and 6-17 of the LTP for the concrete walls and steel liner, respectively.

6.5.3.1 NRC Evaluation of Excavation DCGLs

The NRC staff reviewed the methodology and parameter values used in the development of the excavation scenario DCGLs for concrete basement walls and the steel liner in the containment building. The NRC staff finds that the assumption that the excavated concrete will have the same properties as soil is conservative and would lead to a bounding calculation of the potential dose. The reason for this is that if the if the excavated concrete remains intact, it would be unlikely for an individual to farm in the material. If the future land use does not include farming, the exposure pathways related to farming would not exist, and therefore the total overall dose would be less. The NRC staff concludes that the assumption that residual radioactivity present on the steel liner would release instantly is similarly conservative because it would also lead to a bounding calculation of the potential dose in a given year as compared to a slow release over a long period of time. The NRC staff finds that the use of the DSRs developed for the soil scenario for the dose from the excavated material is appropriate given that both scenarios apply to the same site and NRC staff expects the parameters to be the same. The NRC staff reviewed the calculation of the of the DCGLs from the DSRs and assumed geometry and obtained comparable results. For the reasons described in this paragraph, the NRC staff concludes that the excavation scenario DCGLs are acceptable for evaluating the potential dose from residual radioactivity on the walls and floors if they are excavated.

6.5.4 <u>Basement Wall and Floor DCGLs for ROCs</u>

The licensee developed overall basement wall and floor DCGLs by combining the in situ, drilling spoils, and excavation scenario DCGLs described in the above sections using Equation 6-3 above. The initial suite DCGLs are given in table 6-18 of the LTP. The licensee assigned a dose fraction of 0.05 to the IC radionuclides and calculated basement wall and floor DCGLs for the ROCs (table 6-7). The licensee also calculated OpDCGLs for basement walls and floor using the assumed *a priori* dose fractions provided in table 6-1 above (see table 6-8 below).

Table 6-7: BFM Wall/Floor DCGL for ROC (Adjusted for IC Dose) (Based on Table 6-32 in the LTP)

Radionuclide	BFM Wall/Floor DCGL (pCi/m^2)	
	Auxiliary, Turbine, Circulating Water Tunnels, Intake Structure	Containment
C-14	8.20E+06	8.13E+06
Co-60	3.40E+06	8.21E+05
Cs-137	6.90E+06	2.61E+06
Eu-152	9.62E+06	1.88E+06
Sr-90	8.23E+05	8.46E+05

* Multiply picoCuries (pCi) by 3.7×10^{-2} to obtain Becquerels (Bq)

Table 6-8: Operational DCGLs (OpDCGLs) for Basement Floor/Walls (Adjusted for IC Dose)(Based on Table 5-6 in the LTP)

Radionuclide	BFM Wall/Floor DCGL (pCi/m^2)	
	Auxiliary, Turbine, Circulating Water Tunnels, Intake Structure	Containment
C-14	1.23E+06	1.22E+06
Co-60	5.10E+05	1.23E+05
Cs-137	1.04E+06	3.92E+05
Eu-152	1.44E+06	2.83E+05
Sr-90	1.23E+05	1.27E+05

* Multiply picoCuries (pCi) by 3.7×10^{-2} to obtain Becquerels (Bq)

The licensee stated that SOF based on the Operational DCGL may exceed one without remediation as long as the survey unit passes the Sign test and the mean SOF (when using the Operational DCGL) for the survey unit does not exceed one. The licensee indicated that it would use Equation 5-8 in the LTP to incorporate the dose from these elevated areas. OPPD committed to remediating any area where the Base Case SOF (BcSOF) (i.e., the SOF calculated based on the DCGLs in table 6-7 exceeds one for the basement walls and floors.

6.5.5 <u>NRC Evaluation and Independent Analysis of DCGLs for Backfilled Basements</u>

The NRC staff reviewed the scenarios, parameters, and uncertainty analyses the licensee used to develop the BFM and DCGLs for backfilled basements at the FCS site using section 2.6 of NUREG-1700, which refers to section 5.2 and appendix I of NUREG-1757, Volume 2, Revision 2, and describes, in part, the areas of review pertaining to unrestricted release using site-specific DCGLs. Specifically, the NRC staff verified that the FCS site conditions are adequately addressed in the conceptual model and model assumptions and has confirmed that the licensee has used a mathematical model that is an adequate representation of the conceptual model. The NRC staff finds that the assumptions used to develop the basement DCGLs are consistent with the site conditions described in the LTP. The NRC's conclusion that the parameter values selected are appropriate is dependent on the site conditions being consistent with those described in the LTP. If OPPD finds the site conditions to be different than those assumed in the LTP, or if OPPD finds the source term to change during decommissioning, then the conclusion that these parameters are appropriate might no longer be valid. In that case, as discussed in section 1.3, the licensee would need to evaluate the new information to determine if a change is needed to the methods described in the LTP. Some of the most risk significant parameters in the model include: the configuration of the residual radioactivity in the basement (i.e., the residual radioactivity is a diffuse source term with an approximately uniform distribution, that the buildings are removed to a minimum of 3 ft below grade, a cover depth of 3 ft, minimum wall thickness is as assumed) and the relative ratio of radionuclides, including the IC fractions. For this reason, the NRC staff finds that if new information is found that indicate that these parameter value assumptions are inconsistent with site conditions, then it is particularly important for the licensee to evaluate if a change is needed to the methods in the LTP.

The NRC staff finds that summing the doses from the BFM *in situ*, excavation, and drilling spoils scenarios to establish site-specific basement structure DCGLs is reasonably conservative because all three of these scenarios are not likely to occur simultaneously. As described above, the NRC staff reviewed the calculations of the *in situ*, drilling spoils, and excavation scenario DCGLs and found that they were calculated appropriately. The NRC staff also confirmed the calculations used by the licensee to combine the *in situ*, excavation, and drilling spoils scenarios into the BFR Wall/Floor DCGLs and obtained comparable results. The NRC staff further concludes that the licensee's commitment to remediating any area where the BcSOF is greater than one minimizes the potential for there to be an elevated hot spot in the basements that could cause a significant dose if the area were to be drilled into or excavated. The NRC staff finds that the approach described in the LTP for evaluating elevated areas that exceed the Operational DCGL but are less than the BcDCGL is acceptable because it is consistent with the NRC's guidance in MARSSIM for evaluating the dose from elevated areas.

For the reasons described in this section of the SER, the NRC staff finds reasonable assurance that the site-specific DCGLs to be used in the FSS for backfilled basements that will remain at the site after license termination are adequate to demonstrate compliance with the unrestricted release criteria specified in 10 CFR 20.1402.

6.6 Buried Piping Dose Assessment and DCGLs

6.6.1 Scenarios, Parameters, and Uncertainty Analysis for Buried Piping DCGLs

Section 6.14, "Buried Pipe DCGL," of the LTP discusses the buried piping that OPPD intends to leave onsite at the time of license termination. Buried pipe is defined as pipe that runs through soil and are located outside of basements and structures. The total length and interior surface area of the buried pipe at the time of license termination are listed in table 6-9. OPPD did not survey the buried piping during site characterization. According to the LTP, the licensee will survey the buried pipe as part of "continuing characterization" but does not expect any residual radioactivity to be found beyond the occasional presence of low concentrations near the detection limits.

Table 6-9: Total Length and Surface Area of Buried Piping

Piping Type	Length (m)	Interior Surface Area (m^2)
Storm Drain	955.9	2167.8
Service Water	54.9	12.7
Total	1010.8	2181

6.6.1.1 Scenarios

OPPD determined the buried pipe DCGLs, in units of dpm/100 cm^2, for an excavation scenario and an *in situ* scenario. The excavation scenario assumed OPPD would excavate all buried pipe and place it on the surface. The *in situ* scenario assumed that the pipe would remain buried on the site. OPPD intends to sum the DCGLs for the two scenarios to calculate the buried pipe DCGL. This approach is discussed in further detail below.

The buried pipe excavation scenario assumes that all the buried piping will be excavated after license termination and placed on the ground surface. OPPD assumed all activity associated with the internal surfaces of the pipe would be instantly released and mixed with surface soil over a contiguous area of 2,181 m^2, the total surface area of the buried pipe. Using details provided in section 6.14.1 and appendix A to chapter 6 of the LTP, NRC staff calculated a slightly larger but comparable total internal surface area for the buried pipe. The licensee performed a sensitivity analysis using 0.15 m and 1 m mixing depths associated with the placement of the buried pipes on the surface. Either depth is reasonable given the extensive disturbance to the ground surface that would be required to remove all the buried pipe at one time. Considering a vadose zone thickness of 1.1 m, the unsaturated zone thicknesses for the mixing depths are 0.95 m and 0.1 m, respectively. Results indicate that the highest doses were associated with a mixing thickness of 0.15 m for all ROCs except C-14; a mixing thickness of 1.0 m resulted in higher doses for C-14.

The *in situ* buried pipe scenario assumes that all the buried piping remains in the "as-left" condition, buried in the ground at a depth of 1.1 m, at the time of license termination and all associated activity is immediately released from the interior of the pipes to an adjacent 0.0254 m layer of soil over a contiguous area of 2,167.8 m^2 of subsurface soil. OPPD did not consider the thickness of the pipe or possible impacts the *in situ* pipe may have on the environmental transport of radionuclides released from the pipes. Thus, the source term consists of a 0.0254 m layer of soil located in the saturated zone covered with a 1.1 m thick layer of clean material. The licensee performed sensitivity analyses to validate the performance of the RESRAD non-dispersion groundwater model and to ensure that modeling the source term in the saturated zone is conservative.

6.6.2 Calculating Buried Pipe DCGLs

The licensee used soil concentrations to determine the buried pipe DCGLs since the buried pipe exposure scenarios assume that the concentrations associated with the buried pipes are immediately released to the soil. The licensee calculated a unitized source term (pCi/g per dpm/cm^2) for both the excavation and *in situ* scenarios using Equation 6-17 of the LTP. The unitized soil concentrations are provided in table 6-10. OPPD used RESRAD-ONSITE, Version 7.2, to calculate radionuclide-specific DSRs (mrem/yr per pCi/g), which it combined with the unitized source term values using Equation 6-19 of the LTP to calculate the buried pipe DCGLs for both the excavation and *in situ* scenarios. It then combined the ROC DCGL values for each scenario, resulting in a single conservative BcDCGL.

Table 6-10: Unitized soil concentrations (pCi/g per dpm/cm^2)

Scenario	Pipe Depth (cm)	Soil concentration (pCi/g per dpm/cm^2)
Excavation Scenario	15 cm	0.02
	100 cm	.003
In situ Scenario	2.54 cm	0.12

* Multiply picoCuries (pCi) by 3.7×10^{-2} to obtain Becquerels (Bq)

Table 6-11, "RESRAD-ONSITE, Version 7.2, parameter values for calculating buried pipe DSR values," provides a list of the deterministic parameter values used to calculate the buried pipe DSRs for both the excavation and *in situ* exposure scenarios. The buried pipe excavation scenario uses the same values used to calculate the soil DSR values except for the "Area of Contaminated Zone (m^2)," "Thickness of Contaminated Zone (m)," and "Thickness of Unsaturated Zone (m^2)." The "Length Parallel to Aquifer Flow (m)" value is also changed to reflect the use of the 2,181 m^2 contaminated zone area. The changes in parameter values account for the surface area of the pipes as the scenario assumes that the contaminants are released from the pipes to the surface soil immediately upon being brought to the surface.

Table 6-11 also lists the changes made to the deterministic parameter values for the *in situ* scenario to account for changes to the hydrogeological parameters and K_d values that satisfy the conceptual model assumptions that the buried pipe is fully submerged in the saturated zone.

Table 6-11: RESRAD-ONSITE, Version 7.2, parameter values for calculating buried pipe DSR values

Parameter	Excavation Scenario	In situ Scenario
Area of Contaminated Zone (m^2)	2181	2181
Thickness of Contaminated zone (m)[1]	0.15 m or 1.0 m	0.0254 m
Length Parallel to Aquifer Flow (m)	26	26
Thickness of Unsaturated Zone (m)[2]	0.1 m for 1 m mixing thickness 0.95 m for 0.15 m mixing thickness	
Cover Depth (m)		1.1
Cover Erosion Rate (m/yr)		RESRAD Default PDF
Contaminated Zone Erosion Rate[3]		0
Contaminated Fraction Below the Water Table		1
Contaminated Zone Density (g/cm^3)		1.49
Contaminated Zone Total Porosity		0.45
Contaminated Zone Field Capacity		0.24
Contaminated Zone Hydraulic Conductivity (m/yr)		4350
Contaminated zone deterministic K_d values changed to the 50^{th} percentile of PDF in table 2.13.1 or 2.13.10 of ANL/EVS/TM-14/4		Radionuclide Dependent
Contaminated zone K_d PDFs changed to the distributions in tables 2.13.1 or 2.13.10 of ANL/EVS/TM-14/4.		Radionuclide Dependent PDF

1) Thickness of contaminated zone is the mixing depth
2) The site conceptual model vadose zone thickness is 1.1 m (see section 6.7). The unsaturated zone thickness is 1.1 m minus the mixing depth.
3) Contaminated zone is completely submerged in saturated zone

OPPD summed the excavation scenario and *in situ* scenario DCGLs from table 6-27 of the LTP using Equation 6-20 in the LTP to calculate buried pipe base case DCGLs, a conservative approach since the buried pipes cannot remain in the ground and on the surface simultaneously. OPPD then multiplied the resulting values, listed in table 6-12, by the a priori fraction for buried pipe provided in table 6.1 above and 0.95 to calculate the OpDCGLs for buried pipe. These values are listed in table 6-13 and used in combination with the OpDCGLs for other media of concern to provide reasonable assurance that the summed dose from all media is 0.25 mSv/yr (25 mrem/yr) or less after all FSSs are completed.

Table 6-12: Buried Pipe DCGL (No IC Dose Correction) ROCs

Radionuclide	Excavation Scenario DCGL (dpm/100 cm2)	*In situ* Scenario DCGL (dpm/100 cm2)	Initial Suite DCGL (dpm/100 cm2)
C-14*	3.596E+06	8.874E+06	2.56E+06
Co-60	2.103E+04	1.836E+06	2.08E+04
Cs-137	8.432E+04	1.558E+06	8.00E+04
Eu-152	4.610E+04	2.653E+08	4.61E+04

* Values based on a contaminated zone depth of 1.0 m; the remaining radionuclides are based on a contaminated zone depth of 0.15 m

Table 6-13: Buried Pipe OpDCGLs (Adjusted for IC Dose)

Radionuclide	OpDCGL (dpm/100 cm2)
C-14	3.40E+05
Co-60	2.77E+03
Cs-137	1.06E+04
Eu-152	6.13E+03

6.6.3 NRC Evaluation and Independent Analysis of Buried Piping DCGLs

The NRC staff has determined that the scenarios assumed for buried piping, while not realistic, are conservative. Since it is not physically possible for the entire source term from buried piping to be simultaneously excavated and placed on the surface and remain submerged, adding the excavation and *in situ* DCGL values is conservative. Additionally, since different radionuclides provide doses through different exposure pathways (e.g., groundwater, plant uptake, etc.), it is appropriate to simplify the conceptual model and take the maximum DSR from either the *in situ* 0.15 m unsaturated scenario or the *in situ* 1.0 m saturated scenario for each radionuclide.

Staff also finds it reasonable to consider the RESRAD-ONSITE parameter values used for determining soil DSR values with minimal changes when determining the buried pipe DSR values as the scenario assumes that radioactive material associated with excavated pipes will be immediately released to the soil. As noted above, the licensee performed a sensitivity analysis to evaluate possible impacts from using 0.15 m and 1.0 m mixing depths to assess the dose once the material is brought to the surface. NRC staff considers this approach to be reasonable as it considers mixing depths associated with bringing the buried pipes to the surface as well as surface mixing once the pipes are placed on the surface. In addition, the DCGLs for each mixing depth are compared and the lowest DCGL for each ROC is the value used for compliance.

As part of the RAIs, the NRC staff asked the licensee to clarify the piping surface area calculations, which are ultimately used as the contaminated area considered in the dose assessment. Using the additional information provided in the RAI response, NRC staff calculated a slightly larger surface area, 2183 m^2, then what was provided by the licensee, 2168 m^2, for the storm drain piping. The NRC staff found that this minimal difference did not impact the DSR calculations.

The staff has reviewed the dose modeling analyses for unrestricted release of buried piping as part of the review of the licensee's LTP, using the NUREG-1757, Volume 2, Revision 2. Staff concluded that the licensee has applied an appropriate combination of the conceptual model, exposure scenarios, mathematical models, and input parameters. The licensee has adequately considered the uncertainties inherent in the modeling analysis consistent with 10 CFR 50.82(a)(9)(D) For the reasons described above, the NRC staff have reasonable assurance that the DCGLs to be used for the buried pipe are adequate to demonstrate compliance with the unrestricted release criteria specified in 10 CFR 20.1402.

6.7 Groundwater Dose Approach

The licensee provided a description of the FSS plan and development of pathway dose conversion factors (PDCFs) for the dose contribution from existing groundwater contamination in sections 5.2 and 5.4 of the LTP. Staff will refer to these dose conversion factors as groundwater PDCFs in its review below. Using RESRAD-ONSITE, the licensee estimated the groundwater PDCFs in the LTP, which will be multiplied by the concentration of existing groundwater contamination at the time of the FSS evaluation to obtain a dose. For the FSS, OPPD will compare the estimated dose against a predetermined fraction of the site release criteria that it allotted to existing groundwater contamination.

Staff notes that OPPD treats dose due to existing groundwater contamination[5] separately from assessments of the groundwater pathways that are a component of the scenarios for other end state media, i.e., basement concrete, embedded pipe, buried pipe, and soil. Description and evaluation of the groundwater pathway characteristics OPPD used to calculate DCGLs for other end state media are not addressed here, but rather, are evaluated in sections 6.4, 6.5, 6.6, and 6.10 of the SER.

For FSS planning in table 5-4 of the LTP, the licensee allotted a projected, or a priori, dose fraction of 0.02 to existing groundwater contamination. This fraction of the 0.25 mSv/yr (25 mrem/yr) release criteria in 10 CFR Part 20 is 0.005 mSv/yr (0.5 mrem/yr). OPPD stated this fraction to be the operational DCGL for the existing groundwater medium. The licensee based the dose fractions in table 5-4 of the LTP on site characterization, process knowledge, and extent of planned remediation. For existing groundwater, the characterization used historical results from the groundwater monitoring program as briefly described in section 2.4 of the LTP. The licensee described the historical groundwater contamination as sporadic strontium-90 and tritium levels over the period 2011 through 2018. SER section 6.12.7 provides staff review of historical groundwater data. In section 4.2 of the LTP, the licensee indicated that no remediation actions were necessary or expected for the existing groundwater contamination. The licensee considered the possibility of increased levels of groundwater contamination being identified in the future, but prior to license termination, exceeding that of the historical records. If higher groundwater contamination levels are found in the future, the licensee stated in section 5.2.6.6 of the LTP that the allotted fraction for existing groundwater would increase, with the increase compensated for by a reduction in the fractions for other end state media listed in table 5-4 of the LTP such that the total dose for all media remains below 0.25 mSv/yr (25 mrem/yr). Staff notes that if a significant plume is identified during decommissioning, then ALARA requirements of 10 CFR 20.1402 may lead to a revision of the LTP and associated license amendment.

In section 5.2.6.1.11 of the LTP, the licensee stated that it will calculate the dose contribution from existing groundwater contamination by multiplying the groundwater contamination levels for each ROC by dose conversion factors. The licensee stated in section 5.4.1.10 of the LTP

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[5] The term existing groundwater contamination is a decommissioning term referring to residual radioactivity present in the groundwater phase at the time of the licensee FSS evaluation. It may use historical data or measurements made at the time of the FSS evaluation, as such, it should account for past releases during operation or releases that may occur at the time of dismantlement and demolition.

that it will assess any residual radioactivity via the groundwater monitoring wells installed at the site. In section 5.2.6.1.11 of the LTP, the licensee stated that only positively detected ROCs from the groundwater monitoring program will be included in the dose calculation. The dose conversion factors are calculated by dividing RESRAD output of water-dependent dose by well water radionuclide concentration, as described in Equation 6-21 of the LTP. The RESRAD output and resulting dose conversion factor are reproduced in tTable 6-14. In section 6.18 of the LTP, the licensee described the selection of the ROCs and calculation of dose conversion factors. The latter is based on RESRAD analyses utilizing the embedded pipe scenario. The licensee stated that the scenario used for these calculations does not matter for estimates of dose conversion factors for existing groundwater as long as the contaminated zone is greater than 20,000 m^2 (215,280 ft^2); the licensee correctly pointed out that the RESRAD-ONSITE "results need to include non-negligible well water concentrations" for the calculation of the PDCF. Staff notes that that there are several RESRAD-ONSITE options, including the use of a large area for the contaminated zone, that reduce the possibility that the RESRAD-ONSITE abstraction reduces the well water concentration to negligible values. Equation 6-21 of the LTP illustrated how OPPD calculated the dose conversion factors from the RESRAD output of water-dependent dose at year one divided by well water concentration at year one.

Table 6-14: Dose Conversion Factors for Existing Groundwater ROCs (Reproduced from Table 6-35 of the LTP)

ROCs for Groundwater	Well Water Concentration at 1 Year (pCi/L)	Water Dependent Dose at 1 Year (mrem/yr)	Dose Conversion Factor (mrem/yr per pCi/L)
C-14	1.663E+02	0.45	2.68E-03
Co-60	5.886E+00	0.15	2.52E-02
Cs-137	5.886E+00	0.11	6.86E-02
Eu-152	2.505E+00	0.01	3.63E-03
Sr-90	3.585E+01	3.95	1.10E-01

* Multiply picoCuries (pCi) by 3.7×10^{-2} to obtain Becquerels (Bq)
** Multiply millirem (mrem) by 10^{-5} to obtain Sieverts (Sv)

6.7.1 NRC Evaluation of Groundwater Dose Approach

The NRC staff reviewed the information provided in the FCS LTP pertaining to the licensee's approach for assessing the dose resulting from existing groundwater residual radioactivity remaining at the end of the decommissioning process using NUREG-1757, Volume 2, Revision 2, section 5.2 and appendices F and I. The findings and conclusions of the review result from the evaluation of the licensee's compliance with 10 CFR 20.1402 using the methods described in the FCS LTP. The NRC staff reviewed the assumptions used by the licensee to characterize the facility's existing groundwater residual radioactivity for estimation of dose. The key areas of staff review are the (i) approach for allotting dose for existing groundwater contamination, (ii) basis for groundwater ROCs, (iii) consistency of ROCs with radionuclides measured in the groundwater monitoring program, (iv) approach for establishing the maximum groundwater contamination for the site at the end of decommissioning, and (v) support for the PDCF for groundwater. The staff also reviewed the determination that some of the potential radionuclides are insignificant contributors to dose. These five key areas are discussed below.

For the first key area, the licensee a priori allotted a dose of 0.005 mSv/yr (0.5 mrem/yr) for dose due to existing groundwater contamination with a caveat for the possibility of exceeding this dose. The caveat is that the dose for other media will be proportionally reduced if the allotment for existing groundwater is exceeded due to contamination exceeding historical levels, such as if decommissioning activities lead to a leak or perturbation of solid media contamination that causes increased groundwater contamination. NRC staff finds this plan acceptable because it accounts for the possibility of residual radioactivity results exceeding reported historical results, including possibly elevated values due to key areas 2, 3, and 4.

For the second key area, staff evaluated the ROCs for existing groundwater contamination by reviewing the radionuclides for other media. Based on the description in section 5.2.5 of the LTP, staff confirmed that all ROCs from other media are included in the ROCs for existing groundwater contamination. Staff notes that it is possible that insignificant contributors for other media might be significant contributors for the groundwater medium. For this category of radionuclides, the most likely radionuclides from the initial suite for the site (table 5-1 of the LTP) are those that have both low sorption coefficients and high dose conversion factors for the site-specific exposure scenario pathway. Low sorption coefficients are most likely to desorb from solid media and enter the groundwater system. More details are provided in SER section 6.12.7. In response to RAI TE-6, the licensee provided a plan in section 5.4.1.10 of LTP for 10 percent of groundwater samples to include measurement of all radionuclides in the initial suite listed in table 5.1 of the LTP, which includes both ROCs and insignificant contributors. NRC finds this approach acceptable because the plan includes all ROCs from other media and because data from actual measurements will be available during the FSS to assess insignificant contributors for existing groundwater contamination.

For the third key area, staff compared the groundwater ROCs to the list of measured radionuclides in the groundwater monitoring program. The licensee stated that it would use the results of the groundwater monitoring program as input to the FSS dose estimate for existing groundwater contamination. Staff notes that the list of measurements for groundwater samples in the monitoring program includes tritium measured quarterly and total gamma, Sr-90, Fe-55, and Ni-63 measured at least once a year. The licensee stated that the ROCs for groundwater, however, were C-14, Co-60, Cs-137, Eu-152, and Sr-90. Hence, staff notes that the existing monitoring program does not include a measurement for the beta emitter C-14. Nor is a plan provided by the licensee to address follow-up measurements to delineate the ROC gamma emitters Co-60, Cs-137, and Eu-152, if a positive result is found in a total gamma measurement. OPPD should provide some measure of groundwater ROC concentrations for Co-60, Cs-137, and Eu-152that would be used as input for the FSS calculation of dose for existing groundwater contamination. The NRC staff reviewed OPPD’s groundwater monitoring program and discusses it in SER section 6.12.7. In response to RAI TE-6, the licensee indicated in section 5.4.1.10 of LTP that it will incorporate measurements of all groundwater ROCs into the groundwater monitoring program. NRC finds this plan acceptable because data for each groundwater ROC will be available for the FSS dose assessment.

For the fourth key area, staff evaluated the planned approach the licensee would use to estimate the maximum existing groundwater contamination at the site. Staff notes that the groundwater monitoring network responds in a delayed and dampened manner to releases at the site, particular given the typical situation where the monitoring wells are located at some distance from likely source release areas. In LTP Revision 0, the licensee did not include a plan for assessing trends or a justification for why the well data reflects maximum contamination for the site. In section 2.4.2 of the LTP, the licensee described the historical groundwater contamination as sporadic Sr-90 and tritium over the period 2011 through 2018. Although the term residual radioactivity is used, staff notes that OPPD provided no site-specific quantification of background radioactivity, i.e., no wells identified as background wells, nor any background data collected for groundwater near or at the site. Because the licensee provided no site-specific background data, staff notes that radionuclide concentrations in monitoring wells are assumed to be plant-derived and therefore included in dose calculations. In addition, the licensee reported results below the lower limit of detection in the annual effluent reports. The licensee stated in section 5.2.6.1.11 of the LTP that if OPPD does not positively detect a given ROC in any groundwater sample, then it will assume the dose to be zero. The licensee, however, did not provide the metric for positively detecting an ROC in the groundwater. It provides neither sample-specific uncertainty nor lower limits of detection with the results in the annual effluent reports. In the response to RAI TE2-3 (ML23236A478), the licensee indicated that it would use the MDC to determine detection instead of the critical limit (L_c). Because section 2.4.2 of the LTP stated that the sporadic concentrations of Sr-90 in the shallow aquifer were plant-derived, the draft response to TE2-3 indicated that the next revision of the LTP would state that the Sr-90 in groundwater was a statistical artifact caused by high strontium in the groundwater. Staff notes that OPPD provided no site-specific strontium measurements of the alluvial aquifer, and no justification for using MDC instead of the critical limit, L_c, or defining LLD in terms of a detection decision. NRC guidance in NUREG-1576 (2004, MARLAP guidance) explicitly states that MDC should not be used for the detection decision. Although guidance documents like NUREG-1576 do not contain regulatory requirements, OPPD did not justify using the MDC in this fashion in a way that satisfied the regulatory requirements. Staff notes that laboratory industry standards ANSI N42.23, “Measurement and Associated Instrument Quality Assurance for Radioassay Laboratories” and U.S. Geological Survey “Interpretating and Reporting Radiological Water-Quality Data (USGS, 2008)[6] are consistent with 2004 MARLAP guidance for reporting laboratory results. Additional details are provided in SER section 6.12.7 of staff’s review of the input to estimate the maximum existing groundwater contamination for FSS dose calculation. In response to further licensee and staff discussions of RAIs TE-6 and TE2-3, the licensee revised section 5.4.1.10 of the LTP to state that it will follow the MARLAP guidance for reporting laboratory results. The licensee further stated that it will apply a factor of 2 to address the relationship between well monitoring data and maximum concentration beneath the Deconstruction Area. NRC staff finds the plan for reporting laboratory results provided in section 5.4.1.10 of the LTP acceptable because it is consistent with MARLAP guidance. Staff notes an adequate justification for an alternative approach for the detection decision and reporting of laboratory values would also be acceptable. In addition, staff finds the incorporation of a factor of 2 for the relationship between monitoring well results and groundwater conditions below the Deconstruction Area acceptable because it conservatively compensates for the uncertainty of dispersion and influence of unsaturated zone processes, which is discussed more extensively in SER section 6.12.7.

For the fifth key area, staff evaluated the groundwater PDCF shown in table 6-14, above. As table 6-14 shows, the licensee used groundwater concentration and water-dependent dose output at time equal to one year from RESRAD-ONSITE simulations of the buried pipe media. The licensee selected the buried pipe scenario because it included the water-dependent pathway without being influenced by unsaturated zone processes. The licensee also noted that the source area was larger than the well zone of influence. Staff notes that the PDCF estimate only requires a significant concentration at the well such that the water-dependent pathway dose can be reliably estimated. Due to the ratios of well water concentrations in tTable 6-14, staff inferred that OPPD used the set of deterministic values of sorption coefficient to generate the PDCFs. Staff also notes that the buried pipe media has a thin source, thus the inventory

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[6] USGS, 2008, "Interpreting and Reporting Radiological Water-Quality Data," Techniques and Methods 5-B6, Book 5 Laboratory Analysis, Section B Methods of the National Water Quality Laboratory Chapter 6, U.S. Geological Survey.

exhausts quickly in the 1,000-yr performance period for most radionuclides. Using the buried pipe model and RESRAD-ONSITE, staff could not confirm the PDCF values in table 6-35 of the LTP because it could not reproduce the magnitude of well water concentrations at one year using an unmodified version of the licensee's buried pipe model. Staff notes that when well water concentrations are small, the resulting dose conversion factors may be unstable. A thin source, use of the non-dispersion transport model when wells are much deeper than the contaminated zone, and significantly large sorption coefficient values all combine to produce extremely small well concentrations. Staff's well concentrations are two to four orders of magnitude smaller than those reported in table 6-35 of the LTP. To produce more stable dose response results for each ROC, staff switched to the mass balance model and increased the source concentration. Staff also changed the integration factor in RESRAD-ONSITE to reduce the dissonance of an integrated dose value being compared to a point estimate of well water concentration. With these changes, staff confirmed that PDCFs for C-14, Co-60, Cs-137, and Eu-152 in table 6-35 of the LTP (shown here in table 6-14) are acceptable because staff's calculations show that the licensee's PDCF for these radionuclides are not underestimated. However, staff's calculations indicate that the licensee's Sr-90 PDCF may be slightly underestimated by a factor of approximately two. Because (i) the allotted dose for existing groundwater is a small portion of the dose criteria in 10 CFR 20.1402, (ii) based on Sr-90 historical levels, the increase in dose due to Sr-90 would be a small fraction of a mrem/yr and would lead to an estimate of total dose due to existing groundwater contamination that remains below the amount allotted by the licensee in the LTP, and (iii) the combined uncertainties in measurements and in calculation of PDCFs appear to be larger than the factor of two underestimate of Sr-90 PDCF. Therefore, using a risk-informed approach, staff determines that the difference in the staff's and the licensee's PDCF values is not significant. Therefore, staff finds the PDCFs for existing groundwater contamination are acceptable.

For the reasons described in each of the five key areas of this section, the NRC staff finds reasonable assurance that the approach to assessing existing groundwater dose will be adequate to demonstrate compliance with the unrestricted release criteria specified in 10 CFR 20.1402.

6.8 Above-Grade Structure Dose Assessment

Section 6.17 of the LTP, "Above Ground Building DCGL for ROC," discusses the release criteria that OPPD will apply for the above-grade structures at FCS that will remain post-license termination, which include:

- Training Building,
- FLEX Building,
- Owner Controlled Area Entrance Building,
- 3451 Old Building,
- 3451 New Building, and
- 1251 Control and Switchgear Building.

According to the LTP, these buildings are all Class 3. The licensee indicated that it is using the screening values in table H.1 of NUREG-1757, Volume 2, Revision 2, for all ROCs except for Eu-152. The licensee further stated that the site conditions described in in appendix H of NUREG-1757, Volume 2, Revision 2, for screening criteria apply for these buildings. The licensee noted that table H.1 does not include a screening value for Eu-152, so it is using the "Pcrit 0.9" value from table 5.19 in NUREG/CR-5512, Volume 3 for that isotope. OPPD assigned a value of 0.05 for the IC dose fraction assigned to aboveground buildings. The DCGLs proposed by the licensee for aboveground buildings in the LTP are provided in table 6-15 below.

Table 6-15: Above-ground Building DCGLs for ROCs (based on Table 6-35 in the LTP)

Radionuclide	Above-ground Building DCGL (dpm/100 cm^2)	Above-ground Building DCGL Adjusted for IC Dose (dpm/100 cm^2)
C-14	3.70E+06	3.52E+06
Co-60	7.10E+03	6.75E+03
Cs-137	2.80E+04	2.66E+04
Eu-152	1.27E+04	1.21E+04

6.8.1 NRC Evaluation of Above Grade Structure Screening Values

Section H.1.3 of NUREG-1757, Volume 2, Revision 2, describes the site conditions that should exist for the use of the screening values in table H.1 when demonstrating compliance with the dose criteria in 10 CFR 20, Subpart E, for building surfaces. Provided that these conditions are confirmed to be met in the buildings on-site, the NRC staff finds that the use of the NUREG-1757 screening values in table H.1, for the above-grade buildings that will remain at FCS after license termination is acceptable. The NRC staff also concludes that the use of the NUREG-5512, Volume 3, table 5.19, value for the Eu-152 DCGL for aboveground buildings is acceptable because this concentration is equivalent to the 0.9 quantile for the 10 CFR 20.1402 unrestricted release criteria of 0.25 mSv/yr (25 mrem/yr) (i.e., this concentration corresponded to a dose less than the release criteria in 90% of the probabilistic calculations). For these reasons, the NRC concludes that the DCGL values proposed by the licensee for aboveground buildings are acceptable for demonstrating compliance with the unrestricted release criteria in 10 CFR 20.1402.

6.9 Basement Embedded Pipe Dose Assessment and DCGLs

Section 6.13 of the LTP describes the dose analysis that the licensee performed for the embedded piping in the basement. The licensee indicated that both the Auxiliary Building and Turbine Building basements have embedded pipes, located at the 989- and 971-foot elevations in the Auxiliary Building basement, and at the 990-foot elevation in the Turbine Building. The LTP indicates that OPPD will decontaminate and characterize the Auxiliary Building embedded pipes while it will characterize the Turbine Building embedded pipes as part of continuing characterization. The licensee assigned an IC dose fraction of 0.1 for the embedded pipes in the basement to account for uncertainty in the radionuclide mixtures.

The licensee used the *in situ* scenario described above to evaluate the potential dose from residual radioactivity in the embedded pipes. The licensee used the same parameters for the development of the DCGLs for the embedded pipe scenario as for the basement *in situ* scenario with the exception of parameters related to the physical configuration and size of the contaminated area (i.e., the area of the contaminated zone, length parallel to the aquifer, thickness of the contaminated zone, and cover depth). OPPD used a value of 20,000 m^2 because this area is conservative and maximizes the dose calculated in RESRAD. OPPD modeled the thickness of the contaminated zone as a 1 m thick mixing layer. OPPD used a value of 3.92 m for the cover depth because this is the expected distance between the soil surface and the embedded pipes. OPPD calculated the length parallel to the aquifer based on the square root of the largest basement area. OPPD noted that this parameter does not affect the calculated dose since the source term was modeled as being in the saturated zone. The licensee considered the dose from drilling into the pipes as a less likely but plausible scenario as described in section 6.11 of this SER. The DCGLs developed by the licensee for embedded piping are reproduced in table 6-16 below.

Table 6-16: Basement Embedded Pipe DCGLs for ROCs (Adjusted for IC Dose) (based on Table 6-33 in the LTP)

Radionuclide	Auxiliary Floor 971' Elevation (pCi/m^2)	Auxiliary Floor 989' Elevation (pCi/m^2)	Turbine Floor 990' Elevation (pCi/m^2)
C-14	1.41E+09	1.10E+09	7.57E+08
Co-60	4.83E+09	3.77E+09	2.59E+09
Cs-137	7.53E+09	5.88E+09	4.04E+09
Eu-152	N/A	N/A	4.58E+10
Ni-63	9.39E+10	7.33E+10	N/A
Sr-90	2.04E+08	1.59E+08	1.09E+08

* Multiply picoCuries (pCi) by 3.7×10^{-2} to obtain Becquerels (Bq)

6.9.1 NRC Evaluation of Basement Embedded Pipe Dose Assessment and DCGLs

The NRC staff reviewed the scenario and parameters OPPD used to develop the DCGLs for embedded piping and finds that the use of the *in situ* scenario is acceptable for evaluating the dose from the embedded piping for the compliance scenario since this scenario is consistent with the current physical configuration of the site. As described in section 6.11, the NRC also agrees with the licensee that the embedded piping drilling spoils scenario needs to be evaluated as a less likely but plausible scenario during FSS. The NRC staff finds that the use of the same parameters for the calculation of the DSRs for the embedded pipes as for the basements is appropriate because these pipes are located in the basements and the NRC staff expects the parameter values would therefore be the same. The NRC staff also finds that the parameter values used for modeling the physical configuration of the source term are acceptable because they are consistent with the physical configuration of the buried pipes or are conservative. The NRC staff agrees with the use of the IC fraction of 0.1 but notes that the assumed IC dose fraction is a key assumption that should be confirmed during the licensee's planned characterization of the embedded pipes as is discussed in more detail in section 5.2 above. The NRC staff performed independent calculations of the DCGLs for the basement embedded pipes and obtained comparable results as the licensee. For these reasons, the NRC staff concludes that the DCGL values proposed by the licensee for embedded pipes are acceptable for demonstrating compliance with the unrestricted release criteria in 10 CFR 20.1402.

6.10 Basement Fill Dose Assessment

In the LTP, the licensee indicated that it plans to use 132,000 cubic yards of spoils produced from the excavation of the rail spur area, as fill material after FSS of the structure surfaces and embedded pipe. OPPD excavated the material from a Class 3 impacted area on the FCS site.

To evaluate the potential dose from the fill material, the licensee used the RESRAD code using the same parameter values as for the *in situ* basement scenario (section 6.5.1 of this SER) with a few exceptions. For the fill material dose calculations, it expanded the source term area was expanded to 20,000 m^2 and increased the length parallel to the aquifer flow to 83 m. OPPD assumed the concentration of radioactivity in the fill would be uniform at all depths. The licensee indicated that this was a conservative assumption since it expects the residual radioactivity in the Class 3 area from which the fill originated to be contained in the top 0.15 m surface layer.

Table 6-37 of the LTP provides the full list of fill *in situ* Scenario DCGLs without an IC Dose Correction as well as the Fill material DCGLs for the ROCs corrected for the IC contributor dose. The IC Dose Corrected Fill DCGLs for the ROCs are reproduced below table 6-17. The IC dose fraction for the fill was assumed to be 0.1.

Table 6-17: Fill *in situ* Scenario ROC DCGLs (IC Dose Corrected) (Reproduced from Table 6-37 of the LTP)

Radionuclide	Fill *in situ* DCGL Adjusted for IC Dose (mrem/yr per pCi/g)
C-14	1.29E+01
Co-60	1.59E+01
Cs-137	1.36E+01
Eu-152	5.50E+02

* Multiply picoCuries (pCi) by 3.7×10^{-2} to obtain Becquerels (Bq)
** Multiply millirem (mrem) by 10^{-5} to obtain Sieverts (Sv)

6.10.1 NRC Evaluation of the Basement Fill Dose

The NRC staff reviewed the methodology and parameters used to develop the DCGLs for the basement fill. The NRC staff concludes that it was appropriate to use the same RESRAD parameters to model the fill as for the *in situ* scenario because the same basements are being modeled in both cases. The NRC staff finds that the licensee's use of different parameter values for source term area and the length parallel to aquifer are acceptable since the values selected are consistent with the geometry of the basement fill. The NRC staff reviewed the RESRAD calculations of the DCGLs and performed independent calculations and obtained comparable results. The NRC staff agrees with the use of the IC fraction of 0.1 but notes that the assumed IC dose fraction is a key assumption that should be confirmed during the licensee's planned characterization of the embedded pipes as is discussed in more detail in section 5.2 above. For these reasons, the NRC finds that the use of these DCGLs to determine the dose contribution from the basement fill is acceptable for demonstrating compliance with the unrestricted release criteria in 10 CFR 20.1402.

6.11 Alternate Scenarios: Less Likely but Plausible Exposure Scenarios

As described in section 6.21 of the LTP, the licensee also performed evaluations of the potential dose consequences from two scenarios that it considers to be less likely but plausible. These scenarios were a drilling spoils scenario in which OPPD assumed the source term would be embedded pipe and an offsite processing/recycling of excavated basement concrete and steel liners.

The embedded pipe drilling spoils scenario uses the same assumptions as the drilling spoils scenario described in 6.5.2 of this SER except OPPD assumed the well installer would continue drilling through the floor to a depth of 21.4 m and encounter embedded pipe in the process. As in the drilling spoils scenario, OPPD assumed the cuttings would be mixed and spread over the ground surface. It assumed this well would be drilled at a time of 30 years in the future. OPPD calculated DCGLs for this scenario and found that for some radionuclides, the DCGLs generated using this scenario were lower than the DCGLs generated using the embedded pipe scenario. For these radionuclides, the dose from the embedded pipe drilling spoils scenario would exceed 0.25 mSv/yr (25 mrem/yr) if the radionuclides were present at a concentration equal to the *in situ* embedded pipe DCGLs described in section 6.9. For this reason, in the LTP, OPPD committed to evaluating the dose from the less likely but plausible embedded pipe drilling spoils scenario during the FSS process to ensure that the dose from the embedded pipe drilling spoils scenario is less than 0.25 mSv/yr (25 mrem/yr).

In the offsite processing/recycling of excavated basement concrete and containment steel liner scenario, OPPD assumed a large-scale industrial project would proceed on the site in the future. It assumed a time of 30 years would elapse before this project is undertaken since there are currently no plans to conduct a large-scale excavation on site. OPPD used the dose factors from NUREG-1640 to calculate concentrations of radionuclides that would result in a dose of 0.25 mSv/yr (25 mrem/yr) for this scenario. OPPD reported that the calculated concentrations are all more than the BFM wall/floor DCGLs and therefore the dose from the offsite processing/recycling scenario would be less than the dose from the BFM wall/floor compliance scenario.

6.11.1 <u>NRC Evaluation of the Alternate Scenarios</u>

The NRC staff reviewed the information provided in the FCS LTP pertaining to the licensee's assessment of the potential doses resulting from alternative land uses of the FCS site in the future. The NRC staff conducted this review using section 5.2 of NUREG-1757, Volume 2, Revision 2, which states the following with regard to dose from "less likely but plausible land uses" that may be used at the FCS site:

> If the licensee evaluated scenarios based on reasonably foreseeable land uses, the licensee needs to provide either a quantitative analysis of or a qualitative argument discounting the need to analyze all scenarios generated from the less likely but plausible land uses. The results of these analyses will be used by the staff to evaluate the degree of sensitivity of dose to overall scenario assumptions (and the associated parameter assumptions). The reviewer will consider both the magnitude and time of the peak dose from these scenarios. If peak doses from the less likely but plausible land use scenarios are significant, the licensee would need to provide greater assurance that the scenario is unlikely to occur, especially during the period of peak dose.

The NRC staff finds that the scenarios identified by the licensee as less likely but plausible are consistent with the guidance in NUREG-1757, Volume 2, Revision 2, as alternative land use scenarios. The NRC staff concludes that it is reasonable for the licensee to assume that these scenarios will not occur for at least 30 years since there is no plan for significant drilling or construction at the site at this time. The NRC staff agrees with OPPD’s commitment to analyze the dose from the embedded pipe drilling spoils scenario as part of FSS since the potential dose from this scenario could be above 0.25 mSv/yr (25 mrem/yr) if the pipe concentrations are equal to the DCGL values. The NRC finds that the licensee adequately evaluated the potential dose from the offsite processing/recycling scenario to demonstrate that the dose from this less likely but plausible scenario would not exceed the 0.25 mSv/yr (25 mrem/yr) dose criteria in 20.1402.

6.12 Geology and Hydrology

In accordance with the requirements of 10 CFR 50.82(a)(9)(ii)(A), and consistent with the guidance contained in NUREG-1757, Volume 2, Revision 2, and the NUREG-1700, this section of the SER describes and evaluates the geologic and hydrologic conditions at the FCS site. The SRP guidance indicates that site hydrogeological information should be provided if the licensee proposed site-specific DCGLs. The SRP goes on to state that the hydrogeological information “will likely be required to support the parameters used in the site-specific dose assessment” as described in NUREG-1757, Volume 2, Revision 2. The materials the NRC staff reviewed, and the NRC staff's evaluation of those materials, is based on the information provided in the FCS LTP, supporting documents referenced in the LTP, and the NRC staff’s independent assessment of the FCS LTP, supporting documents, and general literature.

Staff reviewed the FCS hydrogeological information in section 6.2 of the LTP and supporting documents to evaluate the conceptual site model for groundwater flow and transport and its implementation in the dose assessment models that include a groundwater exposure pathway. Groundwater levels and flow beneath the FCS site depend on the (1) river stage that strongly influences groundwater levels under the plant site; (2) hydrogeology of the site; and (3) infiltration and recharge which depend on meteorological conditions. The NRC staff also evaluates the geological and hydrogeological conditions at the FCS to determine whether operations or decommissioning activities have resulted in radiological impacts within the groundwater.

6.12.1 Conceptual Site Model

NUREG-1757 guidance states the importance of describing a hydrologic conceptual site model (CSM) for the flow and transport pathways at a site and how it is implemented in a dose assessment model. The CSM lays out the transport pathways and migration rates of residual radioactivity from source areas to points of exposure.

The licensee described in section 6.2.2 of the LTP a CSM whereby the groundwater conditions at the Fort Calhoun site are driven by bank storage tied to river stage and a vertical contribution from recharge derived from precipitation. For the bank storage hydrologic concept, the groundwater in the saturated zone under the site is in strong communication with the stage level of the Missouri River, i.e., the groundwater flows both towards the river and away from the river depending on whether the river stage is declining or rising.

In section 6.7, the licensee laid out the implementation of the CSM in the dose assessment model that included precipitation-related infiltration leading to water flux through contaminated zones in the unsaturated zone, residual radioactivity carried vertically to the saturated zone, and laterally transport towards the river. The groundwater exposure point is the usage of well water in the resident farmer scenario. The CSM is developed by the ensemble contributions of meteorology and climate, surface water, geology and hydrogeology, groundwater system, groundwater monitoring and ROCs, relevant HSA events (e.g., leaks, spills), and groundwater use, which staff individually evaluates in the following subsections. As such, the adequacy of the hydrologic CSM is evaluated at the end (in section 6.12.10) after assessing site conditions for each of the above components and their contribution to the development of the CSM.

6.12.2 Meteorology and Climate

In Section 1.3.4, “Meteorology and Climatology,” and Section 8.5.1.4, “Climate,” of the LTP the licensee described the weather conditions at the site as cyclically affected by the regional humid climate zone to the east and southeast, the dry zone to the west, and the cold continental polar air from the north. The licensee reported a mean annual temperature of 51.1 ºF (10.6 ºC) with average January and July temperatures of 20.2 and 77.7 ºF (-6.6 and 25.4 ºC), respectively. Section 1.3.4 indicated that the annual precipitation is 0.76 m (29.9 inches) with 75 percent occurring from April through September. OPPD took the value for precipitation from the DSAR [Decommissioning Safety Analysis Report, OPPD, 2017, ML18010A129] and reflects data up to 1990 according to table A.1.1 (attachment to chapter 6). The licensee stated that it reviewed records from the Nebraska State Climate Office for the Eppley Airfield in Omaha for the years 1981 to 2010 for significant differences with the pre-1990 data in the DSAR and did not find any significant differences in several relevant meteorological statistics. In section 1.3.4 of the LTP, the licensee stated that the wind speed and direction varied seasonally. The licensee stated that the annual average wind speed was (3.8 meters per second (m/s) or 8.5 miles per hour (mph) based on the mid-point of the data range presented in the DSAR (OPPD, 2017). In section 8.5.1.4 of the FCS LTP, the licensee reported average wind speed of 4.74 m/s (10.6 mph) and annual precipitation of 0.72 meters (28.5 inches).

For precipitation rate, the dose assessment model described in table A.1.1 of the LTP (Attachment 6-1 to chapter 6) used an annual deterministic value of 0.76 m/yr (29.9 inches/yr) for precipitation based on the DSAR (OPPD, 2017, chapter 2.5). Staff notes that the DSAR reported precipitation annual averages from the North Omaha National Weather Service station for the period 1954-1990 and for the Eppley Airfield in Omaha for 1936-1990; the licensee chose the value from the North Omaha station as input to the RESRAD model. The licensee stated in table A.1.1 of the LTP that records from the Eppley Airfield for the period 1981-2010 were additionally reviewed to ensure no significant changes since 1990. Staff reviewed the reasonableness of using the 1954-1990 North Omaha weather station precipitation data to represent the Fort Calhoun site conditions. Based on data from the Western Region Climate Center (WRCC), which is one of the regional centers chartered to manage data for the National Weather Service, annual precipitation was 0.733 m (28.86 inches) for the Blair, Nebraska, weather station for the period 1893-2006. The WRCC also reported annual average precipitation for overlapping 30-year periods starting from 1961 and ending 2010. The annual average precipitation gradually increased with time from 0.764 to 0.805 m (30.07 to 31.69 inches) for each 30-yr period. The NRC staff concludes that use of 0.76 m (29.6 inches) for the precipitation parameter in the overall dose assessment is reasonable based on comparison with data records from closer stations at the towns of Blair and Fort Calhoun. Further, the NRC concludes that because the annual precipitation amount is a medium priority parameter (see table A.1.1 of the LTP) meaning the small differences in annual precipitation rates noted above are not significant for the dose assessment.

For wind speed, the dose assessment model described in table A.1.1 of the LTP used a stochastic annual average with a uniform distribution between 2.8 and 4.7 m/s (6.3 and 10.6 mph) stated to be based on data in the DSAR (OPPD, 2017). The average value of wind speed in the stochastic sampling of RESRAD-ONSITE simulations would be 13.6 kilometers per hour (km/hr) (8.45 mph). Staff notes that the range cited to the DSAR comes from the 1985-1989 average of 6.3 mph for Fort Calhoun, Nebraska (DSAR table 2.5.1), and the 1936-1990 annual average of 17.1 km/hr (10.6 mph) from Eppley Airfield in Omaha (DSAR table 2.5.2). To address the short record of the Fort Calhoun dataset and the relevancy of the distal location of Omaha to Fort Calhoun, staff compared longer-period averages for the town of Fort Calhoun and for Omaha. A wind speed average of 16.4 km/hr (10.2 mph) for Fort Calhoun, Nebraska, over the period 2004-2022 and a value of 17.7 km/hr (11.0 mph) for Omaha over the period 1955-2022. Table A.1.1 of the LTP indicated that wind speed is a medium priority parameter. Staff notes that there is no mention of the FC site meteorological station data and how it might compare with longer records of nearby stations. Staff finds that the wind speed discrepancy is not important because difference in wind speed is not likely important to dose.

The NRC staff evaluated the information provided by the licensee and concluded that the FCS LTP and supporting documents provide an adequate level of support for meteorological inputs in the overall dose assessment.

6.12.3 Geology

The licensee provided descriptions of the physiography and geology of the site in Section 1.3.3, "Topography," Section 1.3.5, "Geology and Seismology," and Section 6.2.1, "Geology," of the LTP. OPPD cited the DSAR (Defueled Safety Analysis Report, 2017, section 2.6 and appendix F, ML18010A129) and TSSD Services (2017, "Fort Calhoun Station Limited Site Non-Radiological Characterization Survey," ML21271A181) as supporting documentation.

The site is located within the Missouri River bottomland that is a nearly level valley approximately 8 mi (13 km) wide near the site and bounded by bluffs on both sides. The FC site is near the western bluff of the valley. The Missouri River bounds the site on the northeast side (plant-east) and flows from the northwest to the southeast. The U.S. Corps of Engineers moved the river channel eastward in the 1950s and placed revetment structures along the shore. The river stage is nominally at an elevation of 992 ft (302 m) AMSL. The land rises to 1004 ft (306 m) AMSL before dropping slightly to 1000 ft (305 m) AMSL in sloughs on each side of the plant, then rises back up to approximately 1004 ft (306 m) AMSL before dropping off again to 1000 ft (305 m) AMSL further inland. The sloughs reflect an old channel on plant-north and plant south sides that would cross the plant area if not for the leveling to 1004 ft (306 m) AMSL during construction of the buildings. Much of the area within the site boundary is between 1000 and 1004 ft (305 and 306 m) AMSL. There is relatively flat farmland within the site boundary on the plant-north and plant-south sides of the FC property. Further inland, the land surface rises approximately 60 ft to a terrace immediately beyond the rail tracks on the westernmost part of the site, then rises further to the west outside of the site boundary.

In sections 1.3.5 and 6.2.1 of the LTP, the licensee stated that the soils below the site include limestone, dolomite, shale, sandstone, and thin coal layers. The licensee used the term soils in reference to both the bedrock and unconsolidated sediments and distinguished the two by the use of deep and shallow formations, respectively. The site is underlain by a 65- to 75-foot (20- to 23-m) thickness of glacial and fluvial unconsolidated sediments of Pleistocene and recent age. The unconsolidated sediments are underlain Pennsylvanian-aged limestone and shale of the Kansas City Group below the site. The LTP stated that the bedrock interface was relatively flat and the DSAR section 2.6 described 13 feet (4.0 m) of relief. Appendix C of the DSAR (ML18010A129) and section 2.4.2 of the Flood Report (2012, ML21272A219) note that a thin oolitic limestone overlines an aphanitic limestone below the facility. The presence of groundwater flow-enhanced solution cavities in the aphanitic limestone influenced plant construction.

Consistent with the labeling as fluvial and glacial deposits, sections 1.3.5 and 6.2.1 of the LTP described the unconsolidated sediment layers as interstratified and cross-bedded. The upper units include former river deposits including lenses reflective of migrating channels associated with paleo-oxbow deposits. The licensee noted that the paleochannels may lead to preferential flow. The licensee stated that the lower unconsolidated units were likely predominantly Pleistocene glacial and fluvial sediments and included some fine to coarse sands and gravels. The beds or lenses change facies or grade laterally such that OPPD stated that correlation between boreholes across the site was not possible. The upper 20 to 50 feet of unconsolidated sediments are variously described in section 1.3.5 of the LTP as fine-grained sandy clay with silt and in section 6.2.1 of the LTP as fine-grained sand and silt with some clay. The bluffs to the west of the river bottomland contain glacial loess deposits.

The end state subsurface media appropriate for the 1,000-yr performance period also includes backfill. OPPD provided no soil textural description of the construction backfill surrounding buildings during construction, nor the fill to be used for excavations and basement structures during decommissioning. Section 3.4.2 of the LTP described the staging of backfill from the expanded rail spur for later use as fill material for basement structures. Section 5.4.1.5 of the LTP mentioned the potential for using offsite material for fill. The licensee suggested that the same fill material would be used for cover and fill of basements; the cover would comprise the unsaturated zone and the basement fill would be in the saturated zone. None of these invocations of source fill provided a description of the sediment texture needed for selection of sand- or loam-based distributions of sorption coefficients.

Staff finds that the physiography and geology are adequately described for the purpose of supporting the framework embedded in the conceptual site model. However, staff notes that the description of the distribution and texture of near-surface geological media at the site may be inadequate to support the licensee's selection of sorption coefficient distributions. Staff reviews the geological support for sorption coefficients representative of site conditions in the following subsection.

6.12.4 Sorption Coefficients

The licensee selected distributions of sorption coefficients for uncertainty analyses based on textural characteristics of unconsolidated sediments at the site. For uncertainty analyses, OPPD provided ranges of sorption coefficient (K_d) values in table A.1.1 of the LTP. The licensee cited the source of the ranges as the sand, loam, and generic (default values) tables of the Data Collection Handbook (Argonne National Laboratory [ANL], 2015).[7] Similarly, for the soil scenario DCGLs, tables 6-5 and 6-6 of the LTP provided deterministic K_d values based on the Data Collection Handbook tables. For both the uncertainty analyses and the deterministic K_d values for soil DCGLs, the licensee selected values from the loam table for the unsaturated zone and contaminated zone in the RESRAD simulations. OPPD chose the sand K_d values for the

[7] Argonne National Laboratory, 2015, "Data Collection Handbook to Support Modeling Impacts of Radioactive Material in Soil and Building Structures," Argonne National Laboratory, Environmental Sciences Division ANL/EVS/TM-12/4.

saturated zone in all scenarios, including the unsaturated and contaminated zones in the BFM and the contaminated zone for the buried pipe scenario. Because the release model uses an equilibrium desorption-based model, the consequence of the loam assumption for the contaminated zone leads to smaller releases from the contaminated zone, possibly affecting the sensitivity of dose to saturated zone parameters. Staff reviews the use and the consequence of sorption coefficients for the dose assessment model in section 6.4.3 in the SER.

The licensee provided descriptions of the unconsolidated sediments in sections 1.3.5 and 6.2.1 of the LTP. The Defueled Safety Analysis Report (DSAR) and TSSD (2016) were cited as supporting documents for geology in section 1.3.5 of the LTP; instead of TSSD (2016), staff reviewed a similarly titled document TSSD (2017, ML21271A181, Limited Site Non-Radiological Characterization Survey Report). Staff focused its review on the geological characterization of the near-surface material that could be used to support the selection of sorption coefficient values based on the loam table for the contaminated zone and unsaturated zone in the dose assessment.

Various sections of the LTP and cited supporting documents described the upper 20 to 50 feet (6.1 to 15 m) of unconsolidated sediments using different textural terms. These are:

- LTP section 1.3.5 as fine-grained sandy clay with silt
- LTP section 6.2.1 as fine-grained sand and silt with some clay
- DSAR section 2.6 as predominantly silty sands
- TSSD (2017) section 2.2.2 as fine-grained sandy clay with silt

Because these different descriptions do not provide a clear rationale for setting contaminated zone and unsaturated zone sorption coefficients to values corresponding to loam instead of sand, staff reviewed borehole logs in the DSAR, Haley and Aldrich (2021, ML22034A594), TSSD (2021, ML21721A266), and TSSD (2017, ML21271A181). Generally, OPPD drilled boreholes without logging the first approximately 3 m (10 ft), which staff notes is not an unusual practice with disturbed material at the surface, such as construction backfill. Below the uppermost 3 m (10 ft), staff found that the borehole logs exhibit a significant amount of sand and loam consistent with a fluvial environment. Staff further notes that the low gradients and high hydraulic conductivity values at the site indicate the predominant effect of sand layers on groundwater flow at the site (staff reviews hydraulic properties in section 6.12.6 of the SER). Because staff could not find support for loam as the relevant material at the site for setting unsaturated zone and contaminated zone sorption coefficients, staff considered available site-specific soil texture information.

Staff notes that several soil texture classifications are widely used by geological and geotechnical scientists. In Haley and Aldrich (2021, ML22034A594) borehole logs for the monitoring well (MW) series used the Unified Soil Classification System (USCS) and did not include the upper 10 feet (3.0 m) of unconsolidated sediments due to the drilling method. In appendix D of TSSD (2017), logs from Geoprobe corings of the upper 10 to 15 feet (3.0 to 4.6 m) of unconsolidated sediments do not appear to be consistent with the description of sandy clay for the upper 20 to 50 feet (6.1 to 15 ft) in section 2.2.2 of the TSSD (2017). In TSSD (2021, ML21721A266), test borehole logs used qualitative descriptions and tables using the USDA Textural Classification System (TCS) names, though laboratory sample analyses included particle size (sieve) analyses. Neither the USCS nor USDA TCS classification system map cleanly to the soil categories used in sorption coefficient databases, such as that in the Data Collection Handbook (2015) cited in table A.1.1 of the LTP. The compilations of K_d values such as those in the Data Collection Handbook (ANL, 2015) use the following criteria for categories of sand, loam, and clay:

- Sand category was defined as >70% sand-sized particles
- Loam category as even distribution of sand, silt, clay, and up to 80% silt
- Clay category as >=35% clay.

The criteria are based on Sheppard and Thibault (1990),[8] which was the early version of the K_d compilation on which later compilations expanded.

In summary, OPPD provided no description for the near surface [e.g., the upper 10 feet (3.0 m)] unconsolidated sediments in the LTP, which may include disturbed areas, construction backfill material, or natural fluvial sediments. Additionally, staff did not find a geologic (e.g., quantitative or descriptive soil texture) basis for selection of the loam category for K_d ranges for zones at the site where releases to the subsurface may have occurred. Because staff could not find that the sorption coefficient values selected by the licensee were representative of site conditions, particularly for the near-surface material that would comprise much of the contaminated zone and unsaturated zone, staff requested additional information in RAI TE2-11 (ML23151A004).

The licensee stated in its response to RAI TE2-11 (ML23236A478) that it will use laboratory measurements of samples from the site to support the K_d range incorporated in the dose assessment. Previously, the licensee utilized site-specific soil textural information along with data ranges from the loam table in Data Collection Handbook (ANL, 2015) for horizons that contain contaminated soils and for the unsaturated zone. In the response to RAI TE2-11, the licensee indicated that samples for laboratory sorption measurements will include soils from the unsaturated and the saturated zones, basement backfill material, and grout. The licensee indicated that it would provide a report to NRC staff that will include (i) a sample plan that ensures the soils collected and analyzed for sorption characteristics are sufficiently representative of onsite unsaturated and saturated soils, (ii) a description of fill material and grout that will be analyzed for K_d, and (iii) the statistical approach that the licensee will use for selecting K_d values from the sorption tests to apply in reanalysis of DCGLs. For the perspective of site-specific geologic support for sorption coefficients, staff finds the approach planned for supporting sorption coefficient values acceptable because laboratory tests will be performed using samples from the site representative of the relevant media. Staff provides further discussions in section 6.4.3 of the SER on the use and consequence of the sorption coefficient values on dose modeling.

Consistent with staff’s discussion in section 6.4.3 of the SER, the licensee committed in the response to RAI TE2-11 (ML23236A478) to measuring sorption characteristics on relevant site media and ensuring that the contents and results in the licensee’s report pertaining to sorption coefficients meet the assumptions and conclusions in the LTP. If the results of the sorption tests lead the licensee to change the sorption coefficient values used in dose modeling, section 1.3 of the SER describes the process the licensee will use to evaluate whether changes to the sorption coefficient values require a change to the LTP.

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[8] Sheppard, M.I. and D.H. Thibault, 1990, “Default Soil Solid/Liquid Partition Coefficients, Kds for Four Major Soil Types: A Compendium,” Health Physics, Volume 59. Number 4, pp 471-482.

6.12.5 Surface Water

In sections 1.3.2 and 1.3.7 of the LTP, the licensee described the site as being on the west bank of the Missouri River. Other surface water on the site includes two lagoons one-half mile southeast (plant-south) of the protected area and the intermittent Fish Creek on the northwest side of the ISFSI and Switch Yard. In addition, a low-lying swampy paleochannel occurs adjacent to and paralleling the river both plant-north and plant-south. The licensee stated that the U.S. Corps of Engineers armored the shoreline in the vicinity of the site to reduce the possibility of lateral migration of the river towards the facility. In addition, the river stage is regulated by several upstream dams. River stage is an important constraint on groundwater elevations, gradients, and gradient directions under the site and, as such, is discussed further in Section 6.12.6, "Groundwater and Monitoring Network," in this SER.

River stage at the site ranged from a minimum of 300 m (983 ft) AMSL to above flood stage of 306 m (1004 ft) AMSL based on staff assessment of 2007-2021 data provided by the licensee (ML22034A602). Staff calculated an average river stage of 302 m (991 ft) AMSL from the 2007-2021 dataset. Based on USGS approximately monthly measurements over a recent 12-year period at Blair, Nebraska.[9], staff determined that flow rates in the river range from 424.7 to 5,478 m^3/s (15,000 to 203,000 cubic feet per second (cfs)). The lower end of the range is consistent with flow rates during the non-navigation season (DSAR, chapter 2.7). Based on the USGS data, staff noted that the upper end of the range occurred during the 2011 flooding of the site. Staff calculated the median river flow rate as 1,022 m^3/s (36,100 cfs) and the average rate as 1,574 m^3/s (55,600 cfs). The USGS measurements provide an update of the pre-construction ratings curve (stage-discharge curve) in figure 2.7-4 of the DSAR. The flood of 1952 provided an upper constraint on the discharge-stage curve in the DSAR. The flow data illustrates the dilution capability of the Missouri River near the site in the event of potential groundwater residual radioactivity seepage into the river. Staff notes that the annual effluent reports have not identified any offsite migration of residual radioactivity in the groundwater system to the river.

Flooding of portions of the site occurred numerous times before and after construction of the plant. According to the Flood Recovery Action Plan (HDR, 2012), the flood of 2011 was above design basis [306.05 m (1004.1 ft) AMSL] and the FCS remained flooded for approximately three months. Water depths were 0.9 to 1.2 m (3 to 4 ft) and strong currents were observed. Temporary levees and berms protected the main facility buildings, switchyard, and ISFSI. The river stage of the flood of 1952 would have also been above design basis and the discharge was estimated to be approximately twice that of the 2011 flood (HDR, 2012). The 2012 Flood Recovery Action Plan (HDR, 2012) identified other years since construction with various levels of flooding of the site including 1984, 1993, 1997, 2007, and 2010; however, floods during these years may not have reached the design basis flood stage, thus flooding smaller portions of the site. In section 1.3.7 of the LTP, the licensee indicated that 2019 was the latest year that the site experienced flooding and that a barrier around the site was installed in 2020 to reduce the potential for flooding during decommissioning. The barriers are sand- and gravel-filled fabric-lined metal cages that range from 1.5 to 3.0 m (5 to 10 feet) in height. Staff notes that parts of the site and immediate surrounding area are below 305 m (1000 ft) AMSL based on the USGS 7.5-minute topographic map, though the average surface elevation is 306 m (1004 ft) AMSL as indicated in sections 6.2.1 and A.6.1.6 of the LTP.

[9] https://waterdata.usgs.gov/nwis/measurements?site_no=06609100&agency_cd=USGS&format=rdb_expanded, accessed February 22, 2023.

Staff notes that there is no direct surface water input to the RESRAD-ONSITE calculations of DCGLs, but rather surface water considerations support development of the conceptual site model and could influence redistribution of residual radioactivity from events identified in the historical site assessment. For the latter, flooding at the site may lead to redistribution or flushing of near-surface residual radioactivity. Redistribution could potentially include migration of radionuclides incorporated onto solid media (e.g., sediment transport). Flushing refers to radionuclides that are mobile in the water phase and could migrate vertically in the unsaturated zone to the aquifer then laterally to other onsite locations or possibly offsite.

The NRC staff evaluated the information provided by the licensee and concluded that the FCS LTP and supporting documents provide an adequate level of detail regarding the surface water characteristics to support development of a site conceptual model to incorporate into the overall dose assessment.

6.12.6 <u>Groundwater and Monitoring Network</u>

The licensee described the groundwater system in sections 1.3.6 and 6.2.2 of the LTP. Supporting documents cited in these LTP sections include HDR (2012, “Flood Recovery Plan, Revision 4.1,” ML21272A219), RSCS (2008, TSSD #08015, “Review of the Groundwater Protection Program at the Fort Calhoun Nuclear Station, Revision 1,” ML21271A151), TSSD (HSA 2020, “Historical Site Assessment,” ML21271A609), and the DSAR.

6.12.6.1 Groundwater Overview

In section 1.3.6 of the LTP, the licensee indicated that groundwater flow was both towards the Missouri River and away from the river depending on the change in river stage level. The licensee described the conceptual site model as a bank storage system whereby a rising river stage creates gradients away from the river and a receding river stage induces gradient towards the river. The licensee indicated that river stages are relatively high during spring, summer and early fall and river stages are relatively low during late fall and winter.

In section 1.3.6 of the LTP, the licensee cited depths to groundwater ranging from 0.6 to 6.1 m (2 to 20 ft) below ground surface (bgs) and 4.6 to 6.1 m (15 to 20 ft) bgs depending on river stage. It cited different reports for the different ranges. OPPD stated gradients are nearly flat with a gentle slope towards the river at approximately 3.0 m (10 ft) bgs. The licensee also stated in section 1.3.6 of the LTP that water levels varied from 993.7 to 992.4 ft (302.9 to 302.5 m) AMSL, while river stage levels recorded during the same period ranged from 993.2 to 992.4 ft (302.7 to 302.5 m) AMSL. Section 6.2.2 of the LTP clarified that OPPD measured this data in July and August 1966, thus staff notes these conditions and gradients reflect pre-construction conditions. In section 6.2.2 of the LTP, the licensee used the groundwater direction and gradient from the June 16-17, 2020, measurement cycle as the basis supporting flow inputs for RESRAD model. The licensee stated that groundwater depth to water table and contoured groundwater elevation data from June 2020 are consistent with historical data and indicate that groundwater flow is towards the river.

Staff notes that three significant factors make interpretation of groundwater data complex for the purpose of developing a conceptual site model and estimating input to the RESRAD-ONSITE abstracted model for flow and transport. One, over the period 2007-2018, a well pumped 0.76 cubic meters per minute (200 gallons per minute) continuously to supply water for the reverse osmosis plant. The well was located on the northwest corner of the old warehouse slab and said to have a zone of influence on the order of 183 m (600 ft).[10] Two, broken drain lines in the Turbine Building allowed groundwater to enter the Turbine Building basement. Water flow into the building acted as a groundwater sink over an unknown length of time starting in the 1990s when soil cavities were initially identified (Flood Recovery Action Plan, 2012). Third, the measurement cycles, or period of time it takes to measure the water level in all the wells, are generally longer than the period of stable levels of river stage. With wells measured on different days (up to a several weeks) and river stage changing as much as several feet within a day, contouring of water table elevations can be misleading. Staff uses an understanding of the water levels across the site, including the transient behavior, in evaluations of the conceptual site model, the distribution of existing groundwater contamination, and support for hydrological inputs to the dose model in the remainder of section 6.12 of this SER.

6.12.6.2 Monitoring Well Network and Data

In section 2.4.1.2 of the LTP, the licensee indicated that there were 23 onsite monitoring wells, two surface water sites (ISFSI-related), and four storm water headers with routine measurements as part of the Radioactive Groundwater Protection Program (RGPP). The licensee stated that the routine monitoring and sampling followed NEI 07-07 Groundwater Protection Initiative guidance (ML072610036), which is also cited in the ODCM, which the NRC approved (ML21271A600). Figure 2-6 of the LTP illustrated 11 locations with wells; 6 of the locations have a paired set of wells screened at different depths (shallow and deep) and 5 of the locations have a single well screened at shallow depths. The remaining 6 wells of the 23 total indicated to be part of the RGPP are not included in the LTP nor in the 2020 or 2021 Annual Radiological Effluent Release Reports (ARERRs). The HSA (2020) lists 22 wells in its summary of historical information of the groundwater monitoring network.

The RGPP historically had thirteen (13) monitoring wells installed at 8 locations and numbered 1-8. Groundwater depth measurement and sampling began in the last quarter of 2007. The 2008 NEI review of the groundwater protection program (RSCS, 2008, ML21271A151) recommended 3 additional wells at locations on the west and southwest sides of the main plant buildings. The three locations are consistent with wells MW-9, MW-10, and MW-11 locations for which the onset of sampling began in 2008 and 2009. OPPD abandoned these three wells in 2021. OPPD added wells MW-12A/B, MW-13A/B, and MW-14 several years later. According to Terracon (2022), MW-7 replaced by MW15 and MW-4A replaced by MW-16 in September 2021. The location of MW-12A/B on the west side of the electric yard appears to serve as a background well for groundwater chemistry and is in the best position to provide information on groundwater influx from the terrace to the west. However, the groundwater depth information from the well may be affected by proximity to depressions linked to Fish Creek, and no groundwater elevation estimate is possible for comparison to groundwater levels elsewhere at the site since OPPD did not survey the well collar elevation. Staff notes that potential indications are precluded for influx of groundwater from the terraces to the west.

Several onsite locations have paired wells that are screened at different elevations. Monitoring wells named with an “A” or no trailing letter are shallow wells that OPPD installed with screens near and at the water table and to a depth that does not exceed 7.6 m (25 ft), or approximately

[10] Zone of influence of a pumping well implies that groundwater levels are drawn down in a generally conical topology surrounding the wells (i.e. cone of depression), such that monitoring wells withing the zone of influence will exhibit groundwater gradients that differ from the larger scale gradients. The perturbation of the larger scale gradient is dependent on the direction and distance from the pumping well to the monitoring well.

298.4 m (979 ft) AMSL. Groundwater reflected in the “A” series wells likely flows into the river, which has a nominal river bottom elevation of 297.1 m (975 ft) AMSL (section 5.25 of the Flood Recovery Action Plan, 2012). Monitoring wells named with a “B” are deep wells that OPPD screens at a depth interval of approximately 13.7 to 15.2 m (45 to 50 ft) bgs, or an elevation range of 292.3 to 290.8 m (959 to 954 ft) AMSL. The length of the screened portion of the shallow wells is generally 10 ft (3.0 m) and deep wells is 1.5 m (5 ft). Staff notes that the wells emplaced with shallow or deep screens does not mean the there is more than one aquifer in the unconsolidated sediments. The well depths of 7.6 and 15.2 m (25 and 50 ft) bgs do not necessarily correspond to the geologic units exhibited in borehole logs above and below 7.6 m (25 ft) bgs. Staff also notes that hydraulic conductivity measurements from shallow and deep wells in table 5-2 of Haley & Aldrich (2021) were not significantly different. Both shallow and deep values reflect the presence of highly permeable sands. OPPD used data from the paired wells to assess possible vertical flow in the aquifer. Haley & Aldrich (2021) indicated that the vertical gradients range from upward to downward and vary over time at individual well pairs, thus indicating to staff that there is no consistent upward or downward flow in the unconsolidated sediments. Because there is not a consistent pattern, staff concludes that the distribution of vertical flow rate and direction is more likely dependent on changes in river stage levels rather than exchange with a lower aquifer, i.e., the limestone aquifer.

Staff reviewed the groundwater monitoring networks and results of sampling over the period 2007 through 2021 both for the adequacy of the network for decommissioning objectives of detection and for estimation of maximum residual radioactivity in the groundwater. The well locations illustrated in figure 2-6 of the LTP nearly surround the Deconstruction Area. Well locations are situated plant north (1 well), plant-west (2 wells), and plant-south (5 wells). One of the wells counted as being on the plant-south side is the easternmost well (MW-6) and is located on a corner adjacent to the plant-east side of the facility along the river. Whereas there is no well located between the buildings and the river, the MW-6 well is located in a position relative to the facility buildings that would best reflect contaminant flux approaching and subsequently seeping into the river. Staff notes, however, that any seepage into the river would be substantially diluted from the concentrations seen at monitoring wells based on the large flow rates in the Missouri River described in SER section 6.12.5. Because wells have been located on all sides of the facility building, consistent with historical water table data and the bank storage CSM, staff finds that the groundwater monitoring network is adequate for the purpose of detection.

For the objective of estimating maximum residual radioactivity in the groundwater, the licensee stated in the response to RAI TE2-4 that the design of the monitoring well network followed NEI 07-07 guidance. However, staff notes that NEI 07-07 (2007, 2019) guidance objectives are to characterize groundwater conditions and identify radionuclide leaks or plumes (i.e., detection). NEI 07-07(2007, 2019) does not include an objective related to estimating the maximum existing groundwater contamination at the site. Staff notes that potential sources of radionuclides likely originated within the footprint of the facility buildings, but the wells surround the buildings. Although the distance from a potential source to a well outside the building footprint is not large, some amount of dispersion occurs. This means the concentration below the buildings would be some amount larger than what is recorded in the monitoring wells. Because it is not practical to locate wells near likely source release areas due to interference with operations or decommissioning activities, a licensee may propose a method for estimating the maximum concentration if no wells are located near potential sources. OPPD’s approach is discussed in SER section 6.12.7.

For the groundwater direction and gradient, staff reviewed the historical water level data from the groundwater monitoring network. This subsection reviews the data, and the next subsection reviews the estimate of the gradient. Staff analyzed the groundwater data covering 2007 through 2021 provided by OPPD on January 13, 2022 (ML22034A602), supplemented by data from the Flood Recovery Action Plan (2012, ML21272A219) and Haley & Aldrich (2021, ML22034A594). Staff analysis of the dataset indicated that groundwater flowed inland approximately a third of the time, mounded under the facility a third of the time, and flowed towards the river a third of the time. This is consistent with the licensee's description of a bank storage conceptual site model. The licensee also indicated that well pumping on the plant-west side of the facility to supply water for the Reverse Osmosis Plant had a zone of influence of approximately 183 m (600 ft). Staff notes that this cone of depression would have perturbed groundwater flow directions and gradients at the site. In addition, leakage of groundwater into the Turbine Building from broken pipes over a couple decade period also perturbed the groundwater flow direction, especially during higher groundwater levels. Both the pumping well and the leakage into the Turbine Building perturb the groundwater levels nonuniformly across the site, and therefore staff notes that data from the monitoring network would not be representative of future groundwater levels and flow rates to be incorporated into the dose assessment model.

Because of the bank storage CSM, staff's assessment of the groundwater water table data includes both measurements at wells and river stage. Staff notes that river stage measurements at the site provide another groundwater datapoint for understanding the flow system and provide confidence in gradient estimates. Staff notes that the practice of including concurrent site measurements of river stage made during the cycle of water level measurements at wells appeared to be discontinued once decommissioning started in 2016. In the data provided by OPPD on January 13, 2022 (ML22034A602), staff noted that early water level measurements exhibited a distinct and apparently site-specific river stage level recorded for each well. In addition, Terracon (2022) provided an adjustment factor of -0.64 m (-2.11 ft) for translation of Blair, Nebraska, river gage data to FC site river stage conditions for instances when site-specific river stage data was not collected. There was no indication when this translation may have been applied. For data after 2016, the same river stage level was recorded for all wells even if (i) measurements were made on different days or weeks, and (ii) continuously gaged sites at Decatur, Blair, or Omaha indicated a rapidly changing river stage. After 2019, river stage was no longer recorded. For groundwater measurement cycles after decommissioning started in 2016, the licensee reported a river stage value, though it did not appear to represent concurrent site-based values. As an illustration of how strongly the river stage affects groundwater conditions across the site, staff determined that the average river stage elevation during the period 2007-2021 falls in the middle of the range of averages for all wells using only wells with greater than 45 measurements (this only excludes wells with 6 or fewer measurements). The range of average water levels for all the wells is approximately one foot. Yet, staff notes, the range of groundwater levels and the range of river staff span more than 6.1 m (20 ft).

The relevance of staff's observations of OPPD's perturbation of groundwater levels (well pumping and leakage into Turbine Building) and the river stage data is that representative long-term value for groundwater gradient should be taken (i) from the period when no pumping was occurring to support the reverse osmosis plant (exclude 2007-2018), (ii) when groundwater was not flowing into the Turbine building, which is pre-1993 and post-2013 based on RAI TE2-5 response (ML23236A478), and (iii) when staff could better interpret comparisons of well water table levels between wells because concurrent river stage measurements were collected to support the assumption of a constant boundary condition. Because the first two items perturb the groundwater levels and the third item only improves reliability in gradient estimates, the first two items take precedence for estimating future groundwater levels for the site. Therefore, staff focused on data from the post-2018 period. Because the licensee did not perform synoptic measurements[11] for a measurement cycle (i.e., measure all the wells in a short time period such as in one day), staff notes that the standard approach of contouring all the site data to support an estimate of gradient is not appropriate. Instead, staff implemented an alternative approach that only compared well-pairs with measurements made within several hours on the same day. Quantitative estimates from the limited data set using only post-2018 data and well-pairs with approximately synoptic measurements, combined with qualitative assessments that the gradients at the site are generally low because of the highly conductive fluvial sand layers, should provide NRC staff sufficient basis to review the licensee's estimate of the long-term groundwater gradient in the next subsection.

6.12.6.3 Review of Inputs for Hydraulic Conditions and Properties

Staff reviewed the dose model inputs for water table elevation, groundwater gradient, and hydraulic conductivity. In RESRAD-ONSITE, the water table elevation sets the thickness of the unsaturated zone. The magnitude and gradient direction of groundwater support estimates of groundwater flux; the flux is a product of magnitude of the groundwater gradient and the saturated hydraulic conductivity. Staff notes that the direction of groundwater flow, whether inland or towards the river in the bank storage CSM, is not important in the RESRAD-ONSITE calculations because OPPD defined the rectangular contaminated zone as approximately the footprint of all the buildings, which is perpendicular to the river.[12] Staff notes that the appropriate input for gradient in the dose models should represent the post-license termination conditions, thus should be based site conditions unperturbed by pumping or leaks into buildings, in addition to being based on synoptic monitoring well data..

In section 6.2.2 of the LTP, the licensee used the groundwater direction and gradient from the June 16-17, 2020, measurement cycle as the basis supporting flow inputs for RESRAD model. The licensee stated that groundwater depth to water table and contoured groundwater elevation data are consistent with historical data and are flowing towards the river. According to table A.1.1 of the LTP, the licensee selected a value of 0.00084 for the groundwater horizontal gradient for all simulations based on a single measurement cycle from June 2020. In Section 3.4 of Terracon's (2022), "Hydrogeologic Assessment and Conceptual Site Model Report" (ML23346A152, attachment 4), horizontal gradients based on shallow wells ranged from 0.0006 to 0.002 and deeper wells ranged from 0.00082 to 0.001. According to table A.1.1 of the LTP, the licensee used values of 4,350 and 34.4 m/yr (14,270 and 113 ft/yr) for the hydraulic conductivity of the saturated zone and unsaturated zone, respectively.

For the water table elevation, the license stated that typical depths to the water table fell in the range of 3.0 to 4.6 m (10 to 15 ft) bgs based on monitoring data. In the dose models for two different DCGLs, the licensee set the unsaturated zone thickness to different values; a third dose model (buried pipe) neglects the unsaturated zone. The licensee conservatively set the unsaturated zone thickness in the dose model for the soil DCGL to 0.1 m (0.33 ft) for the soil

[11] Synoptic refers to conditions at a particular time. In the Fort Calhoun situation, a change in the river stage during a measurement cycle is reflected in water levels for sets of wells measured on different days. Stated otherwise, the difference between water levels in wells measured one day is much smaller than the difference in water levels for sets of wells measured on different days. Staff analysis of continuous measurements at upstream (Decatur) and downstream (Omaha) river gages indicates that river stage can change several feet over a one-day period.
[12] In RESRAD-ONSITE, the groundwater flow direction is only specified relative to the length and width of the contaminated zone. The actual flow direction at the site is not directly an input in RESRAD-ONSITE.

DCGL (table A.1.1 of the LTP), which corresponds to a water table depth of 1.1 m (3.6 ft) bgs when combined with a 1-m (3.3-ft) thick cover. For the BFM *in situ* DCGL, however, OPPD set the water table to the elevation of the upper part of the wall that would be left in place so that there is no unsaturated zone within the subsurface structure. Because neglecting or setting the unsaturated zone to a small thickness is conservative for dose, staff finds the implementation of a thin unsaturated zone acceptable for use in the dose assessment modeling. Staff notes that neglecting (or reducing the thickness of) the unsaturated zone is conservative because unsaturated zone processes in RESRAD-ONSITE can only delay and reduce radionuclide migration, which reduces maximum dose during the 1,000-yr performance period. Therefore, staff finds that the water elevation selected for determination of the unsaturated zone thickness is acceptable because neglecting or the use of a small unsaturated zone in the dose models is conservative for dose.

For the groundwater gradient, the licensee stated that the gradient based on June 16-17, 2020, groundwater conditions represented average conditions in both the shallow and deep portions of the aquifer. To confirm the licensee's determination, staff's assessed average gradients at the site by considering groundwater data from periods of time expected to reflect long term flow conditions at the site, i.e., for the 1,000-yr performance period. Staff first reviewed the pre-construction data of July and August 1966 in the DSAR and notes that the data is limited to supporting a nearly flat water table. Staff notes that there is no spatial or temporal information associated with the 1966 data, just a range of groundwater conditions across the site and a range of river stage values. Therefore, the DSAR information only qualitatively supports the nature of the groundwater table as having a low gradient, i.e., the gradient cannot be quantified from the information provided in the DSAR. Staff next interpreted groundwater level data from the groundwater monitoring program. Based on staff's review in section 6.12.6.2 of the SER, staff selected well-pairs with groundwater level measurements made after 2018 for estimating gradients reflective of long-term conditions for the site. Staff estimated groundwater gradients ranging from 0.00008 to 0.00187 using well pairs from different sides of the site with water level measurements made on the same day. From 11 gradient estimates derived from quarterly data from 2019 through 2021, staff calculated an average gradient equal to 0.00092, which covered all quarters of the year thus covering different seasons. Staff's calculated gradient values are close to and bracket the value of 0.00084 from June 16-17, 2020, that the licensee selected to use for deterministic RESRAD simulations. Because the licensee's estimate falls within staff's estimate of a range of gradients representative of long-term conditions, staff finds the licensee's estimate of gradient acceptable for use as site-specific input for the dose models. In addition, the licensee indicated in the response to RAI TE2-12 (ML23236A478) that DCGL values were not sensitive to gradient. For example, changes in parameter ranges by a factor of 10 for gradient in the RESRAD BFM *in situ* model had no effect on dose. Therefore, staff finds the input for gradient acceptable for use in the dose models because (i) the values are based on site-specific information, and (ii) dose model results were not sensitive to a reasonable range of gradient values.

Hydraulic conductivity is the second important parameter for groundwater flow. In section 6.2.2, the licensee stated that site measurements of hydraulic conductivity based on slug tests ranged from 9.4e10-3 to 1.8e-2 centimeters per second (9,730 to 18,620 ft/yr) with an average of 4,352 m/yr (14,280 ft/yr). For the unsaturated zone, the licensee reported the average hydraulic conductivity of 34.4 m/y (113 ft/yr) based on falling head tests in boreholes. Staff notes that slug tests are appropriate for estimating hydraulic conductivity in the saturated zone and falling head tests are appropriate for the unsaturated zone. According to table A.1.1 of the LTP, the licensee used values of 4,350 and 34.4 m/yr (14,270 and 113 ft/yr) for the hydraulic conductivity of the saturated zone and unsaturated zone, respectively. The licensee indicated in the response to

RAI TE2-12 (ML23236A478) that DCGL values were not sensitive to hydraulic conductivity. For example, changes in parameter ranges by a factor of 10 for hydraulic conductivity in the RESRAD BFM *in situ* model had no effect on dose. Therefore, staff finds the hydraulic conductivity inputs acceptable for use in the dose models because (i) the values are based on site-specific information, and (ii) dose model results were not sensitive to a reasonable range of hydraulic conductivity values.

Recharge is the third important hydrologic parameter that is important for groundwater flow at the site. Recharge to the saturated zone is calculated in RESRAD-ONSITE based on several input parameters, including precipitation rate, irrigation rate, evapotranspiration coefficient, and runoff coefficient. The licensee provided input values for these parameters in table A.1.1 of the LTP. Using these input values and equations for recharge from RESRAD-ONSITE, staff calculates a value of 0.0613 m/yr (2.4 inch/yr) recharge. As a percentage of precipitation, this recharge rate as a fraction of precipitation rate is 8 percent, which staff notes is larger than a representative percentage of approximately 5 percent[13]typical of the climate and near-surface hydrogeology of the region. Staff finds that the recharge rate, and thus the parameters used to estimate that rate, are acceptable for use in the dose models because the licensee (i) used site-specific data to derive the input values, (ii) those input values led to a recharge rate that is larger than typical for the region, and (iii) a larger recharge rate is conservative for dose.

For the reasons stated above, the NRC staff finds the licensee's site-specific hydrogeological parameter values acceptable for use in the development of a site-specific dose assessment model.

6.12.7 Groundwater Radionuclides of Concern and Historical Measurements

In section 6.18 of the LTP, the licensee indicated that C-14, Co-60, Cs-137, Eu-152, and Sr-90 were ROCs for potentially existing groundwater contamination. In section 5.2.6.1.11 of the LTP, the licensee stated that it would include only positively detected ROCs from the groundwater monitoring program in the dose calculation for existing groundwater contamination. In section 4.2 of the LTP, the licensee stated that characterization survey results and historical survey data indicated that groundwater residual radioactivity was minimal, and thus no remediation actions were necessary.

In section 2.4.2 of the LTP, the licensee provided a summary of historical results for tritium, Sr-90, and total gamma in groundwater for the period 2011 through 2018. For tritium, the licensee reported levels between 8.25 and 15.39 Bq/L (223 and 416 pCi/L) in well MW-6 on three occasions, twice in 2014 and once in 2018. The licensee indicated that (i) these values are near the lower limit of detection (LLD), (ii) tritium was not detected in all other monitoring wells, and (iii) no onsite activities would have triggered an increase in tritium levels. The licensee suggested that these tritium values are potentially false positives. For Sr-90, the licensee reported minor, sporadic detections up to 0.033 Bq/L (0.9 pCi/L) in wells monitoring the shallow groundwater in the Deconstruction Area. OPPD suggested that a small release had occurred in the shallow aquifer at the site in section 2.4.2 of the LTP. However, in response to RAIs TE2-2 and TE2-3 (ML23236A478), the licensee provided an analysis showing the Sr-90 in groundwater was a sporadic statistical artifact caused by large uncertainty due interference caused by high strontium levels or that it was background Sr-90 unrelated to plant operations. For gamma-emitting radionuclides, the licensee stated that it had identified none in groundwater

[13] Szilagyi, J. and J. Jozsa, 2013, MODIS-Aided Statewide Net Groundwater-Recharge Estimation in Nebraska, Ground Water Volume 51, Number 5, pp.735-744.

samples based on total gamma measurements. Staff notes that the gamma-emitting groundwater ROCs include Co-60, Cs-137, and Eu-152.

Based on table III.9 of the ARERRs, staff notes that tritium, Fe-55, Ni-63, Sr-90, and total gamma are measured in the groundwater monitoring program. OPPD reported results for total gamma, Ni-63, and Fe-55 as zero for the period from 2007 through 2022, where it recorded results as zero for results below lower limit of detection (LLD). For tritium and Sr-90, for which the licensee reported non-zero results as noted in the previous paragraph, staff includes a discussion and review below of results reported in the ARERRs in section 6.12.7.4 of the SER.

In the following five subsections, staff (i) compares the list of radionuclides historically measured in the groundwater program with the selected list of ROCs for the FSS, (ii) reviews how the licensee reports groundwater results, (iii) reviews the possibility of background radionuclide concentrations, (iv) reviews the distribution and trend of groundwater monitoring results for the period 2007 through 2022, and finally, (v) provides staff's summary and conclusion for the plan for existing groundwater contamination for the FSS.

6.12.7.1 Groundwater ROCs

The licensee described the selection of groundwater ROCs based on RESRAD modeling using the embedded pipe scenario that is summarized in section 6.18 of the LTP. It listed the groundwater ROCs as C-14, Co-60, Cs-137, Eu-152, and Sr-90. Based on data summaries in the ARERRs and the ODCM, the licensee measured tritium, Fe-55, Ni-63, Sr-90, and total gamma in the groundwater monitoring program during the period 2007 through 2022. Staff notes in the existing groundwater monitoring program, (i) C-14 is not measured and would not be found because it is a beta emitter, (ii) Co-60, Cs-137, and Eu-152 are gamma emitters that are not directly measured, however, because total gamma is measured, a plan for handling non-zero total gamma results should be in place for further analyses, and (iii) Sr-90 is directly measured. Staff notes that the FSS without measurements of the ROCs would not be able to demonstrate compliance with termination dose criteria.

In response to RAI TE-6, the licensee stated in section 5.4.1.10 of LTP, Revision 1, that it will modify the groundwater monitoring program to include measurements of all ROCs for existing groundwater contamination listed in table 6-35 of the LTP. The licensee indicated that monitoring will be in accordance with site procedures, which amongst other requirements, require quarterly sampling. The licensee stated that trends in the data will be analyzed when assessing values for input to the FSS. Staff notes that at the time of FSS, a sufficient amount of data will be needed to adequately demonstrate a stable or declining level of residual radioactivity, if ROCs are detected. In addition, the licensee stated that ten (10) percent or one sample per measurement cycle will include measurement of the full suite of radionuclides listed in table 5-2 of the LTP. Staff finds the modified groundwater program described in section 5.4.1.10 of the LTP acceptable for FSS because the plan will provide actual measurements for FSS input to dose models.

6.12.7.2 Evaluation of Reporting of Radionuclide Results

OPPD reported radiological results from onsite well water samples in the groundwater monitoring program in the ARERRs. Table III.9 in each ARERR provided a summary of results listing non-zero values for results above LLD and zero for results below LLD. OPPD did not provide the values and basis for sample specific LLD in the ARERRs. In the summary of 2021 ARERR, section 6.8, the licensee stated that results below minimum detectable activity (MDA), but above LLD, are provided only for their assessment of trends. In the response to RAI TE2-3 (ML23236A478), the licensee indicated that MDA is used for the decision on the occurrence of Sr-90 in the groundwater (i.e., the detection decision). Also, in response to RAI TE2-3, the licensee indicated that it defined LLD based on laboratory measurement of uncertainty, specifically two standard deviations.

Guidance for radiological measurements is provided in NUREG-1576, Multi-Agency Radiological Laboratory Analytical Protocols Manual (MARLAP). NRC regulatory guides for effluent measurements—RG 1.21, "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste" and RG 4.1, "Radiological Environmental Monitoring for Nuclear Power Plants"—both cite MARLAP for approaches and equations for laboratory radiological analyses. In addition, laboratory industry standards ANSI N42.23, "Measurement and Associated Instrument Quality Assurance for Radioassay Laboratories" and USGS "Interpretating and Reporting Radiological Water-Quality Data (USGS, 2008) are consistent with 2004 MARLAP guidance for reporting laboratory results. MARLAP uses the term MDC, which for purposes here is interchangeable with MDA. To avoid confusion, the NRC staff will use MDC for the remainder of this discussion, even when the licensee used MDA. In chapter 20 of NUREG-1576, MARLAP defines the MDC as "an estimate of the true concentration of analyte required to give a specified high probability that the measured response will be greater than the critical value." If the β value for Type II error is set to 5 percent, then the MDC would be the value of the radionuclide concentration for which there is a 95 percent probability of being above the critical, or threshold, value for the measurement. In other words, the definition of MDC presupposes that an appropriate detection threshold has already been defined. MARLAP indicates that the detection threshold, or critical value, is determined using blank samples and statistical analysis of the uncertainty derived from the distribution of signals from the measurement process used to measure those blanks. MARLAP explicitly states that the detection decision should not be made with the MDC, but rather by comparison with the critical, or threshold, limit. According to MARLAP, the lower limit of detection (LLD) is an older and often misused concept that would be conceptually associated with threshold or critical limit.

The licensee resolved RAI TE2-3 by expanding Section 5.4.1.10 for Revision 1 of the LTP. The licensee stated that MARLAP will be followed, including the utilization of the critical value for the detection decision in accordance with MARLAP. In addition, the licensee stated that the laboratory method uncertainty will be set to a 95 percent confidence level for the action level of MDC divided by three (i.e., the uncertainty level will be a fraction of the MDC value). Following MARLAP guidance, staff notes that the detection decision is based on the critical level, below which the result can be reported as zero concentration. Results between the critical level and MDC may not be sufficiently precisely quantified without further refinement of the laboratory method. In this case, staff notes that a conservative approach that avoids additional laboratory measurements is to set the concentration to the MDC value. Laboratory results above the MDC value are statistically valid results without additional assessment. Staff finds the licensee's plan for reporting laboratory results for groundwater samples acceptable because the licensee will report measurement result in a manner that is consistent both with industry standards and NRC guidance.

6.12.7.3 Background Concentrations

Staff considered the possibility that Sr-90 results from the groundwater monitoring program can be attributed to background Sr-90 beneath and surrounding the site. However, OPPD did not develop or implement a program to characterize background concentration in the groundwater system for use in the FSS. Instead, OPPD cited the FSAR (1970) for groundwater and river sample results for conditions prior to construction of the power plant in the response to RAI TE2-3 (ML23236A478). OPPD indicated that the Sr-90 results found in recent groundwater samples (2007-2022) may be tied to the pre-construction Sr-90 levels, and as such, should be considered background Sr-90. The FSAR (1970) reported an average of 0.004 Bq/L (0.1 pCi/L) Sr-90 from eleven wells within a 4-mile (6.4-km) radius of the FCS. The FSAR (1970) also reported an average of 0.048 Bq/L (1.3 pCi/L) Sr-90 for six regional surface water locations, including surface water at Desota National Wildlife Refuge and river samples ranging from Council Bluffs, Iowa, to Omaha, Nebraska. Since these samples appear to represent regional values for Sr-90, and no nuclear facilities were operating in the area, staff concludes that the average pre-construction Sr-90 results likely represent the signal from radioactive fallout from nuclear weapons testing in the mid-century. With a half-life of 29 years, staff notes that any Sr-90 fallout from nuclear weapons testing would be significantly reduced from the pre-construction values noted above during decommissioning at the FCS site.

Staff also considered the potential for high strontium values to interfere with and contribute to high uncertainties in Sr-90 measurements. In the response to RAI TE2-3 (ML23236A478), the licensee suggested that the high uncertainty in laboratory results was due to interference from high strontium in the groundwater, which led to the sporadic Sr-90 values. In the response, OPPD provided a map of the country from a USGS website that showed high strontium in aquifers of Nebraska. Nebraska and surrounding states in the Midwest are known to be an area with high strontium concentrations in aquifers due to dissolution of calcite in the limestone bedrock (Musgrove, 2021)[14]. However, staff notes that site-specific values of strontium concentrations in the river or groundwater are not reported in the ARERRs or AREORs. The FSAR (1970, ML210128J794) reported no strontium results for groundwater but included one measurement of strontium for river water, which was <0.5 mg/L. Also, staff note that the groundwater monitoring wells are screened in the shallow groundwater of the alluvial aquifer, which consists of fluvial deposits of sands and silty sands that staff notes may not be significantly influenced by the groundwater chemistry from the regional limestone bedrock based on the vertical gradients that are discussed in section 6.12.6.2.

Based on the above discussion, staff cannot reasonably conclude that the sporadic Sr-90 results for groundwater are either background Sr-90 or statistical artifacts due to strontium interference. Staff notes that the LTP was not revised to include the information provided in the response to RAI TE2-3 (ML23236A478) on background Sr-90 and strontium. Therefore, staff finds it appropriate that the licensee will not rely on reducing groundwater concentrations by subtracting a background radionuclide concentration, nor will the licensee claim interference in laboratory analyses by high strontium levels in groundwater samples.

6.12.7.4 Evaluation of Historical Results

Staff reviewed the tritium and Sr-90 concentrations for the period from 2007 through 2021 for all wells reported in table III.9 of the ARERRs. For tritium results from the period from 2007 through 2021, OPPD reported 17 values above LLD at five different wells, all on the plant-south side of the protected area. The maximum was 19.24 Bq/L (520 pCi/L) tritium at well MW-5A in 2010. Ten (10) of the 17 results above LLD were at the well closest to the river, MW-6. Staff notes that data from the ARERRs for the period from 2011 through 2018 indicate 8 occurrences above LLD at five different wells, whereas the summary in section 2.4.2 of the LTP reported three results above LLD and only at well MW-6. Regardless, staff notes that the values of tritium are

[14] Musgrove, M., 2021. The occurrence and distribution of strontium in U.S. groundwater, Applied Geochemistry 126, 104867.

low, and that tritium is both naturally occurring and potentially plant-derived. Also, the only tritium result above LLD over the past two years (2020-2021) would be a factor of approximately 100 times lower than the EPA derived concentration limit for drinking water standards for beta emitters. Using an MDC value for tritium in a dose calculation would still result in an extremely small dose. Hence, staff finds that neglecting tritium as an ROC in the FSS is acceptable, unless future measurements during decommissioning result in significantly larger values of tritium in the groundwater. If significantly larger levels of tritium are found during decommissioning, the LTP would need to be revised to include a pathway dose conversion factor (PDCF) for tritium because none is included in the LTP. PDCFs for groundwater are discussed and reviewed in section 6.7 of the SER.

Staff also reviewed the Sr-90 concentrations for the period from 2007 through 2022 based on data provided in table III.9 in the ARERRs. Several staff observations about the Sr-90 data are:

- The number of Sr-90 results above LLD for all wells is
    - Seventy-two (72) results over 15+ years during the period from 2007 though 2022
    - Thirty-six (36) results over 8 years during the period from 2011 through 2018, which is the period summarized in section 2.4.2 of the LTP
    - Fourteen (14) results over the last 3 years, the period from 2020 through 2022
- Maximum and average Sr-90 of results above LLD
    - For all years, 0.104 and 0.024 Bq/L (2.81 and 0.642 pCi/L), respectively
    - For 2011 through 2018, 0.033 and 0.020 Bq/L (0.890 and 0.535 pCi/L), respectively
    - For last 3 years (2019 through 2021), 0.033 and 0.026 Bq/L (0.878 and 0.710 pCi/L), respectively
- Spatial distribution:
    - Results above LLD have occurred on all sides of the Deconstruction Area
    - The five wells with the most results above LLD, in decreasing order, are MW-6, MW-3A, MW-4A, MW-5A, and MW-7, which represent both the plant-south and plant-north sides of the Deconstruction Area
    - These five wells comprise 83 percent of the Sr-90 results above LLD
    - The two closest wells to the buildings, MW-6 and MW-3A, exhibit the most results above LLD
- Four values of Sr-90 recorded at wells MW-12A and MW-12B in 2010 and 2012; wells MW-12A/B are located southwest of the Switch Yard – far outside of the Deconstruction Area, but near Fish Creek
- Nearly all the positive Sr-90 results are from shallow wells. However, Sr-90 results above LLD occurred in deep wells twice; once each in wells MW-2B and MW-12B in 2017 and 2010, respectively
- Only years 2008, 2015, and 2016 had no Sr-90 results above LLD
- Quarter 3 (typically sampled in August) registered the highest percentage of Sr-90 results above LLD, Quarter 1 (winter) had the lowest percentage
- The number of Sr-90 results above LLD in the three quarters following the 2011 flood is unremarkable compared to other periods.

Staff notes that interpretation of trends and averages is not straightforward due to the complex sampling strategy. The sampling strategy consisted of monitoring wells with Sr-90 measurements made either four, three, two, or one quarter per year. In addition, those strategies changed over time dependent on the particular monitoring well. For wells with at least one non-zero Sr-90 value, only wells MW-6 and MW-3A had continuous quarterly measurements since the fourth quarter of 2007. Staff notes that the description of the sampling strategy is solely for the purpose of interpreting results and is not intended to support a review of the strategy.

Figure 6-1 illustrates the range of Sr-90 values and distribution over time for monitoring wells for results above LLD. A notable feature in Figure 6-1 is the gap in positive Sr-90 results in 2015 and 2016. This is a period when peak river levels in spring and summer were lower than for other years, though there are other years with low peak river stages. Another notable feature is the decrease in the number of positive Sr-90 results for the periods from 2010-2014 and 2017-2021. Staff notes, however, that the decrease may be misleading due to the evolving sampling strategy over time. Specifically, the total number of samples analyzed decreased between the two periods by more than half, but the percentage of Sr-90 results above LLD significantly increased for the later period. To illustrate this point further, there were eight instances when the only measurement made at a well during a year resulted in a Sr-90 value above LLD during the period 2017-2021, whereas there were none during the 2010-2014 period. Therefore, staff cannot discount the possibility that the trend in the number of occurrences of Sr-90 above LLD is caused by the change in sampling strategy rather than a decreased presence of Sr-90 in groundwater. The year with the highest percentage of Sr-90 results is 2020 when 46 percent of the samples were above LLD compared to the period 2010 through 2014 when 22 percent were above LLD. Whereas this increased percentage could reflect an improved sampling strategy, the change in strategy may have also missed possible measurement events exhibiting Sr-90 in the groundwater. Therefore, the number of wells with non-zero Sr-90 levels each quarter or each year should not be compared with other quarters or years.

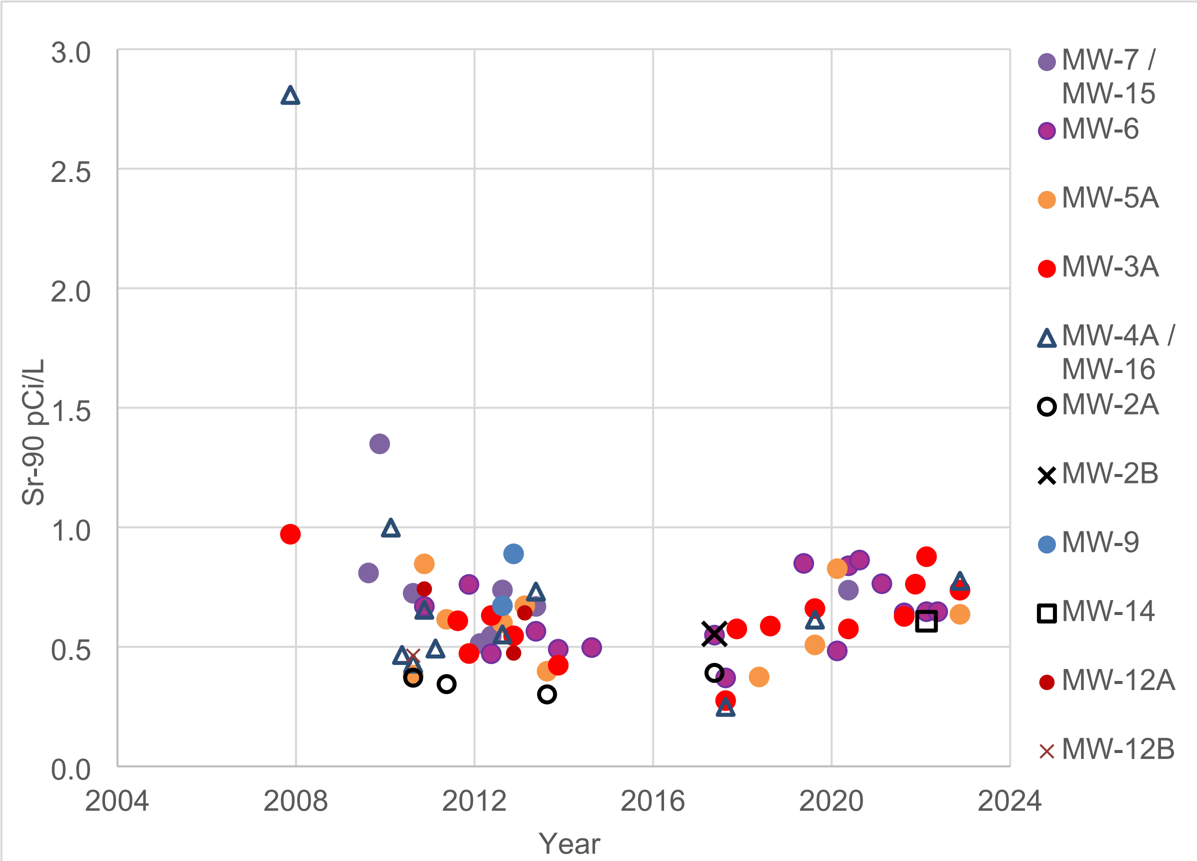

**Figure 6-1*:*	Strontium-90 Results in Groundwater. Wells in the Legend Are Ordered from Plant-North Clockwise to Plant-Northwest Outside the Deconstruction Area.**

Based on the Sr-90 observations, staff cannot discount the possibility of a persistent source in the Deconstruction Area of unknown location and magnitude. In SER section 6.12.8, staff noted at least one HSA incident where spent fuel pool water leaked to the environment; spent fuel pool water could contain Sr-90. Staff notes that the spatial and temporal pattern of Sr-90 results appears to be consistent with a contaminated source area in the unsaturated zone that is activated during higher river stages. Staff notes that the percentage of positive Sr-90 results is inversely correlated with the seasonal river stage. The licensee set sampling cycles to quarterly for the purpose of capturing the seasonal changes. Staff notes that quarters 2 and 3 are periods of high river stage, and these are the quarters with the highest percentage of Sr-90 results above LLD. Spatially, more of the positive Sr-90 results (above LLD) occur at wells closest to the facility with occurrences on all sides. Consistent with the bank storage conceptual site model, the residual radioactivity would migrate in different directions dependent on prior river stage level. Prior to 2018, groundwater directions were also influenced by onsite pumping and influx into the Turbine Building related to broken drain lines (see SER sections 6.12.5 and 6.12.8). During the pre-2018 period, staff assessment of the historical groundwater data indicated gradients approximately equally divided between plant-west (inland), mounded under the site, and plant-east (towards the river). Post-2018, the groundwater gradient direction was more often towards the river. Staff analysis indicated that wells closest to the buildings and furthest from the influence of the pre-2018 pumping well are the wells with the most positive Sr-90 results. Since the groundwater directions varied due to pumping and to changes in river stage, staff cannot use the concept of upgradient wells to represent background or discount plant-derived contamination.

Staff considered the licensee's plan presented in LTP Revision 0 to use concentration values from wells surrounding the facility as direct input into the dose calculation for existing groundwater contamination. Based on the locations of wells in the monitoring network, staff notes that wells may be as much as several hundred feet from a source in the footprint of the buildings or as little as tens of feet, depending on the location of the source and direction the contaminant migrates. Given the distance to the monitoring wells from the potential source area (or areas) within the footprint of the buildings (also described as the Deconstruction Area), the maximum concentration of contamination is likely higher beneath the buildings compared to that at the monitoring wells. Therefore, staff notes that use of the monitoring well sample results directly will under-represent the maximum concentration of an ROC at the site. Staff considered that the time scale over which confirmation of groundwater contamination migration would be reflected at wells outside the footprint of the buildings from releases within the footprint of the facility buildings. Considering the CSM and the magnitude and timing of changes both in monitoring well data and river stage, staff notes that the time scale for migration appears to be less than quarterly, and certainly less than yearly.[15] The hydrological properties and the temporal aspect of the sporadic Sr-90 results are both consistent with this short time scale for migration. Staff notes that FSS groundwater input requires a trend analysis that supports stable or decreasing concentration values for ROCs. Based on site hydrogeologic flow characteristics and the pattern of Sr-90 historical results, staff considers at least two years of quarterly data obtained after the last licensed material is removed from site and the last of the land or subsurface disturbances have occurred as sufficient for estimating radionuclide concentrations for input to dose models at FFS for most sites. Staff considers a two-year period sufficient because both the hydrological flow characteristics operate on and the pattern historical Sr-90 results vary on a much shorter time scale than two years, e.g., any trends in the data would factor in seasonal variations.

Staff also considered the relationship between Sr-90 in soils and in groundwater. This relationship is the underlying basis for sorption coefficients. The undefined potential source area could include Sr-90 in sediments or on subsurface portions of building structures. The licensee does not include Sr-90 in the list of ROCs for soil DCGLs but included it in section 5.2.5 of the LTP Revision 0 as an insignificant contributor. The licensee set Sr-90 to be an ROC for DCGLs for walls and floors. Staff notes that measurements of Sr-90 in subsurface soil samples could be used to infer a source area beneath the buildings capable of leading to contaminated groundwater. With a low sorption coefficient for strontium, staff notes that Sr-90 would readily desorb and migrate with vertical or horizontal groundwater flow in a variably saturated zone. Since these subsurface soil areas are inaccessible during characterization, the licensee does not yet have results for subsurface soil Sr-90. Because OPPD designated Sr-90 an insignificant contributor for soils, the licensee planned to measure Sr-90 and other HTDs in 10 percent of continuing characterization and FSS samples. To address NRC's RAI TE2-2 (ML23151A004), the licensee stated that Sr-90 will be measured in all continuing characterization soil samples in the Deconstruction Area. This statement is reflected in section 5.2.5 of the LTP.

Staff notes that a coherency check on Sr-90 levels in subsurface soil samples and groundwater concentration levels can be made using sorption coefficient for strontium, which assumes

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[15] Staff acknowledges that the time scale for migration to any well will differ based on the distance from the source and the relevant gradient magnitude and hydraulic and transport properties along that pathway.

equilibrium of strontium between solid and water phases. Staff notes that the equilibrium assumption is valid for the saturated zone (neglecting kinetic effects) but likely overestimates groundwater concentrations for variably saturated zones. This check can be performed for any radionuclide, but the coherency check is most appropriate for Sr-90 because of its mobility and large pathway dose conversion factor. Because continuing characterization or FSS soils are not yet available, staff expects to perform its coherency check at the time of the FSS using the largest concentrations in subsurface soils either from continuing characterization, pre-remediation, or FSS sampling. In lieu of Sr-90 soils levels becoming available, staff used the soil Sr-90 value listed in the initial suite of radionuclides for soil DCGLs from the LTP (see SER section 6.4.3 for review of soil DCGLs) to provide context as a bounding value to use in staff's estimate Sr-90 in groundwater. The required input includes the Sr-90 K_d value of 168 L/kg for contaminated zone and unsaturated zone for soil DCGLs from table 6-5 of the LTP and the soil concentrations of 0.411 and 0.064 Bq/L (11.1 and 1.73 pCi/g) from table 6-7 of the LTP for 0.15 m and 1.0 m thicknesses for the contaminated zone, respectively. With the K_d for loam sediments and the soil concentrations, staff calculated the corresponding groundwater concentrations of Sr-90 of 2.44 and 0.37 Bq/L (66 and 10 pCi/L), respectively. Using the K_d value of 22 L/kg for sand sediments from table 6-6 of the LTP, staff calculated corresponding groundwater concentrations of 2.85 and 18.39 Bq/L (77 and 497 pCi/L), respectively. Note that these are concentrations in the groundwater that neglect the effect of processes in the unsaturated zone, which if included, would lead to a decrease in the concentrations reaching the saturated zone. Staff's estimates of groundwater concentrations associated with residual radioactivity in soil samples are provided as examples only. If continuing characterization or FSS soil sample concentrations match the value for Sr-90 listed in table 6-5 of the LTP, the allotted dose for groundwater would be exceeded based on the value of the PDCF the licensee provided in table 6-35 of the LTP for Sr-90 for existing groundwater contamination. Staff note that consideration of unsaturated zone processes could be used to justify lower expected groundwater concentrations. Otherwise, further investigations of areas with significant soil Sr-90 concentrations or a compensating decrease in the allotment for other DCGLs as described in section 5.2.6.6 of the LTP may be needed to meet total dose requirements.

For staff's calculations in the previous paragraph, staff used sorption coefficient values listed in tables 6-5 and 6-6 of the LTP. With the dependency on sorption coefficient values, the estimates of groundwater concentrations in the previous paragraph may change as measurements of site-specific sorption coefficients become available. In response to RAI TE2-11 (ML23236A478), the licensee indicated that measurements of sorption characteristics will be performed on samples representative of unsaturated and saturated soils from the site. If the measured values of sorption coefficients differ from those listed in tables 6-5 and 6-6 of the LTP, staff notes that the same process as described at the end of the previous paragraph would be applicable.

6.12.7.5 Summary and Conclusion

Using information from LTP Revision 0, staff reviewed the groundwater monitoring program that OPPD will use to generate input for the FSS dose for existing groundwater contamination. Staff could not determine if the allotted dose of 0.005 mSv/yr (0.5 mrem/yr) for existing groundwater contamination was acceptable, primarily because:

- Historical measurements and the existing groundwater monitoring program did not include all the ROCs needed for the FSS
- OPPD did not define the LLD, and did not provide values for sample-specific LLD and MDC in the ARERRs or in the LTP Revision 0

- Staff could not determine how the licensee defined the detection decision for laboratory analyses in the ARERRs or LTP, nor how reported measurements of Sr-90 that were statistically above the laboratory detection limit with a high probability would be treated in the FSS
- Although the Sr-90 levels in groundwater are low, staff could not discount the possibility that the Sr-90 in the groundwater may be derived from the plant rather than representing background, and :
  - Based on the spatial and temporal distribution of Sr-90 results, staff could not discount a source in the variably saturated zone considering the fluctuating water table related to the bank storage conceptual site model, onsite pumping, and the groundwater sink caused by influx into the basement of the Turbine Building
  - Staff noted that the spatial and temporal distribution of Sr-90 results did not appear to be consistent with a plume in the saturated zone below the site or any ongoing leak
  - Staff noted that the Sr-90 results were not consistent with pre-construction sources of background contamination, such as the pulse of radioactive fallout from nuclear bomb testing, or with the present-day concentrations in the river
  - The licensee based planned input for ROC concentration for the FSS on results from maximum concentrations in wells; however, staff noted that concentrations at wells located some distance away from possible source areas are lower than that near a source area, thus likely under-representing the maximum concentration at the site.

The LTP incorporated changes to section 5.4.1.10, compared to the information provided in LTP Revision 0, that addresses the summary bullets above. In section 5.4.1.10 of the LTP, the licensee indicated that the groundwater monitoring program will include monitoring of all groundwater ROCs in accordance with site procedures for the monitoring program. Staff notes that the site procedures include monitoring groundwater wells quarterly and concludes that monitoring for two years after OPPD removed licensed material from the site will be sufficient to demonstrate any trends (e.g., stable or declining concentrations) in support of the dose estimate for FSS. Considering that not all the ROCs have been measured historically, staff concludes that a two-year period is sufficient to identify trends that may be affected by seasonal hydrologic conditions. In addition, the licensee stated that it will monitor 10 percent or one sample per groundwater measurement cycle for the full suite of radionuclides listed in table 5-2 of the LTP using a judgmental selection process. Staff finds that a judgmental selection process will be adequate for sampling to support quantification of insignificant contributors of 10 percent of allotted groundwater dose because judgmental selection is conservative compared to random selection. Furthermore, the licensee indicated that it will define the term “positively identified” used in section 5.2.6.1.11 of the LTP consistently with the NUREG-1576 MARLAP guidance discussion on the statistical basis for laboratory results. Specifically, to be consistent with MARLAP, the licensee stated it will use the laboratory-reported sample-specific critical value (L_c) for a detection decision, or justification provided for an alternative approach to MARLAP guidance. For results between sample L_c and MDC, also following MARLAP, OPPD will use the MDC value directly as a conservative input to the FSS dose calculation or perform additional laboratory analyses to better quantify radionuclide levels. To account for the potential increase in concentrations beneath the footprint of the radiologically impacted buildings compared to the locations of monitoring wells, the licensee stated that it will use a factor of two increase in concentrations for input to the FSS dose calculation for existing groundwater contamination.

Because of all the changes incorporated into the LTP as described in the previous paragraph, staff finds with reasonable assurance that OPPD will adequately characterize groundwater contamination, and not underestimate it, for the FSS dose calculation for existing groundwater contamination.

6.12.8 Liquid Radiological Spills, Leaks, and Releases

In section 2.4 of the LTP, the licensee described the ongoing monitoring programs for surface water and groundwater, both onsite and offsite. The NRC staff reviewed the results from the onsite program in SER section 6.12.7 regarding sporadic observations of tritium and Sr-90 in the groundwater. In section 2.4 of the LTP, the licensee did not identify any radiological spill, leak, or release events in the HSA that it linked to the groundwater system.

In section 2.1.4.1 of the LTP, the licensee described the Safety Injection Refueling Water Tank (SIRWT) overflow incident in 1984. The license stated that the event was significant in that the main steam and condensate, feedwater, and blowdown systems were contaminated by reactor coolant. OPPD confirmed primary to secondary leakage by interviews. The license further stated:

> "The SIRWT over flowed to outside railroad siding door. The SIRWT overflow event also caused an overflow of the Spent Fuel Pool. Most of the Spent Fuel Pool water was confined inside the Auxiliary Building; however, some did escape to the adjacent outdoors area. The DA area affected by the SIRWT event was excavated immediately following the event until analysis confirmed the contamination had been removed. The excavated soil was disposed of as radioactive waste."

In addition, the licensee stated that records "providing evidence of environmental contamination resulting from the primary to secondary leaks were not found." Staff notes that excavation of surface soils would remove sorbing radionuclides, but that mobile radionuclides may have infiltrated further down beyond the excavated volume of soil and entered the groundwater system. The SIRWT incident is included in table 4 of the HSA (ML21271A609). One other incident in table 4 of the HSA involved spent fuel pool water. In 2005, increased rates of leakage out of the liner of the spent fuel pool were detected. OPPD provided no further information about containment of this leakage. Staff notes that Sr-90 could be present in spent fuel pool water.

table 4 of the HSA identified other radiological incidents within buildings or rooms, including numerous ones involving water leakage or intrusion into buildings. Staff notes that the occurrence of wall or floor cracks and potentially leaky wall joints was not described, though table 4 listed one instance of a cracked floor (Room 16) in 2014 with potential for inflow and outflow between the building and the natural environment. In section 4.2.2 of the HSA, the licensee noted that residual contamination on equipment and impermeable surfaces potentially could be mobilized and infiltrate to the groundwater. However, the licensee indicated that it immediately isolated and cleaned minor releases from equipment and impermeable surfaces according to procedures, suggesting that the minor releases were unlikely to affect groundwater. In the HSA, the licensee also stated that there were no known unauthorized releases of non-radiological contaminants to the environment.

In section 2.1.4.1 of the LTP, the licensee described the measures taken during the lead up to the 2011 flood at the site. Because of warnings from the U.S. Army Corps of Engineers prior to river stage reaching flood levels, earthen berms were constructed using soil from the site. Following the flood, OPPD surveyed the sandbags from inside the protected area for release. OPPD found detectable Cs-137 and disposed of the sandbags as radioactive waste. Staff notes that the presence of Cs-137 in the sandbags indicates either, the soil used in the earthen dam was contaminated or that mobilization of Cs-137 occurred at the site during the flood. In addition, the sanitary lagoons were also flooded during the 2011 flood event, which is of note because Cs-137 and several other gamma emitters had been identified in the lagoon sludge. Because flooding was one of the possible explanations for the presence of Cs-137 in the earthen dam, staff notes the potential that Cs-137 and other mobile (i.e., small sorption coefficient) radionuclides may have entered the groundwater system during recession of the floods.

Staff noted in the 2012 “Flood Recovery Action Plan 4.1” (ML21272A219) that it first identified broken drain lines and cavities in the sediments in the 1990s. The Flooding Report expressed concern that the cavities in the sediments below the building floor may have been enlarged during the 2011 flood event leading to structural concerns for the building. Groundwater flow and sediment transport occurred into the Turbine building basement with the broken drain lines as the entry pathway. Whereas the Flood Report discussed structural and geotechnical aspects important for the flood recovery planning, it did not provide information on other potential consequences of broken drain lines. Staff is concerned that the broken lines have not been sealed or plugged, liquids draining along the lines to the sump may have contained residual radioactivity, and those liquids may have been released to the unconsolidated sediments below the Turbine Building leading subsurface contamination of sediments and groundwater. In the August 10, 2023, draft response to RAI TE2-5, the licensee provided information indicating that no likely residual radioactivity was released to the natural environment. The information included: (i) the function of the drain lines, (ii) when and how it sealed or grouted the drains, (iii) description of liquids carried in the broken drain lines and any summary sampling or characterization of those liquids in the lines or destination sump, (iv) continuing characterization surveys of drain lines and sampling of sediments near the broken drain lines, and (v) where OPPD will document the characterization (e.g., continuing characterization report and FSS release report) with reference to the broken drain lines.

As described above, staff notes that several incidents could have leaked plant-derived radionuclides to the subsurface. Consistent with the guidance in NUREG-1757 (Volume 2, Revision 2, appendix F), characterization and continuing characterization assessments are warranted based on the HSA-identified events and conditions at the site that potentially led to releases to the subsurface. The incident potentially associated with Sr-90 levels in the groundwater, which staff reviewed in section 6.12.7.4 of the SER, is the SWIRT incident. OPPD stated that the remediation included excavation of soil contaminated by liquid release that likely contained Sr-90 and possibly other radionuclides. For the SWIRT incident, staff cannot discount that the liquid release to the surface and subsurface soil may also have led to liquid migration of more mobile radionuclides such as Sr-90 or tritium to the groundwater.

6.12.9 <u>Water Use</u>

In sections 6.2.4 and 8.6.3.3 of the LTP, the licensee described use of river and groundwater in the area surrounding the site. The licensee stated that water from the river is used for municipal and domestic water supply with the closest upstream intake 1.6 km (1 mile) north and the closest downstream intake 32 km (20 miles) south. The licensee stated that groundwater in the area has a variety of uses, including domestic, public water supply, irrigation or livestock, and commercial or industrial uses. The licensee surveyed wells in river valley over an area 4.0 km (2.5 miles) upstream to 7.2 km (4.5 miles) downstream and generally within 3.2 km (2 miles) of the river. The licensee stated that:

- Thirty-one (31) domestic wells were at depths of 39.6 to 77.7 m (130 to 255 ft) bgs to shale bedrock and at depths 10.4 to 85.0 m (34 to 279 ft) bgs to limestone bedrock
- Twelve (12) irrigation wells ranged in depth from 18.3 to 33.5 m (60 to 110 ft) bgs and had average water levels of 2.6 m (8.5 ft) bgs
- The one livestock well at 44.5 m (148 ft) bgs
- Five (5) public water supply wells with depths up to 30.5 m (100 ft) bgs and water levels ranging from 4.3 to 11 m (14 to 36 ft) bgs; all 5 public water supply wells are located on the east side of the river in the recreation area and wildlife refuge.

For water use, staff determined that well depth and pumping rate are the important parameters related to water use. The licensee used site-specific information for depth of well screen and pumping rate for the resident farmer scenario. OPPD implemented the nondispersion transport model in RESRAD-ONSITE, therefore, dilution of groundwater residual radioactivity in the well bore is calculated by the code and would be incorporated into the calculated dose. In RESRAD-ONSITE, the input parameters of depth of well screen and pumping rate are used to calculate well bore dilution. In addition, well depth is also used in the RESRAD-ONSITE calculation of residual radioactivity in well drilling spoils for the inadvertent intruder scenario; residual radioactivity in drilling spoils is used in RESRAD-ONSITE to calculate dose.

Staff first evaluates the licensee's selection of values for well depths. In table A.1.1 of the LTP, the licensee provided the range of well screen depths sampled for the general uncertainty analysis using a linear distribution. OPPD set the pumping rate to a deterministic value that was stated to be conservative. For simulations of most of the DCGLs, OPPD set well depth (bottom of well screen) to 21.4 m (70.2 ft). The exception is for the BFM *in situ* DCGL groundwater pathway that used 4 m (13 ft) reflective the thickness of the contaminated zone and depth to basement floors. OPPD based the deterministic value for well depth on a survey of nearby wells. In section A.6.1.4 of the LTP, the licensee created a cumulative probability distribution for well depths using 36 of the 43 identified domestic and irrigation wells near the site. In Haley & Aldrich (2021), the bottom of the well screens in the 43 domestic and irrigation wells ranged from 7.0 to 89.0 m (23 to 292 ft) bgs. Staff confirmed that OPPD included the shallowest 8 wells of the 43 wells to create the distribution of well depths. The licensee selected the value associated with the 25th percentile of the well depth distribution for use as a deterministic parameter value in RESRAD simulations for all media except BFM. In RESRAD-ONSITE code, the well is assumed to be screened from the maximum well depth to the water table. Well screen depth is negatively correlated with dose, but OPPD provided no basis for using the 25th percentile value as a conservative input. To evaluate the adequacy of the licensee using the 25th percentile value, staff calculated the dilution for the case when an overly conservative well depth is selected. Using the buried pipe scenario, staff compared well water concentrations when using the licensee value of 21.4 m (70 ft) to concentrations using a staff-determined minimum well depth. More dilution in the well bore would be expected when using deeper wells for a shallow contaminated zone compared to using a shallower well depth. Considering water table fluctuations under the site, staff selected a minimum well depth of 7 m (23 ft) based on the practical aspect of requiring a well be in the saturated zone during all seasons and all years. Using the buried pipe model, staff found higher concentrations for the shallower well by a factor between 2 and 3 depending on the specific ROC and associated sorption coefficient. From the RESRAD-ONSITE simulation results, staff found a dilution factor of 0.02155 for the licensee's selected well depth of 21.4 m (70 ft) compared to the dilution factor of 0.03107 for the overly conservative minimum well depth of 7 m (23 ft). Whereas both of those dilution factors indicate substantial credit from dilution, staff determined that the difference between the values would not lead to a significant change in the final dose estimate. Staff finds the licensee's selection of well depth values acceptable because the licensee used site-specific well depths and staff determined that the use of a 25th percentile did not lead to a significant change in final dose compared to an overly conservative choice,

Staff next evaluates the licensee's selection of values for pumping rate. For well pumping rate, the licensee indicated in sections 6.4.2 and 6.11.3 of the LTP that for the soil and BFM *in situ* models, respectively, it determined the well concentrations were insensitive to pumping rate over a range reflective of residential farmer, farmer with irrigation, recreational, or industrial pumping rates. The licensee calculated a pump rate of 4,550 m^3/yr (160,682 ft^3/yr) using the approach described in section A.10 of NUREG-6697 Attachment C and used an irrigation rate for Nebraska from NUREG-5512 Volume 3.[16] In section A.6.1.5 of the LTP, the licensee described the approach for the calculation of pumping rate. The licensee tabulated three cases that had different amounts of irrigated land. The licensee selected the case with 22,000 m^2 (5.44 acres) of irrigated land, which, by an order of magnitude, is the largest water use contributing to the resident farmer scenario. The uses were irrigation, drinking water for family of 4, household use for a family of 4, and meat and dairy cow water ingestion. Staff notes that including a large irrigation area in the resident farmer scenario leads to large pumping rates that could lead to lower water well concentrations due to increased dilution. However, the licensee provided sensitivity results indicating that well concentrations were not sensitive to the difference in the two scenarios. Therefore, staffs finds that the pumping rate used for dose calculations is acceptable.

6.12.10 NRC Staff Conclusions for Hydrology and Groundwater

Section 6.12 of the SER reviews the site characteristics provided in the FCS LTP that are important for the development of a site conceptual groundwater flow model and parameter inputs for the dose assessment model. The NRC staff finds that the licensee's geological and hydrogeological site characterization is sufficiently detailed to fulfill the site characterization requirement of 10 CFR 50.82(a)(9)(ii)(A) because it is consistent with NUREG-1757, Consolidated Decommissioning Guidance, and the NUREG-1700. The NRC staff finds that the hydrology and groundwater description for the FCS is adequate because observations and measurements taken at the site are generally consistent with the conceptual site model. The observations and measurements include those staff reviewed in subsections of section 6.12 of the SER on meteorology and climate, geology, surface water, groundwater monitoring network and historical measurements, and water use, which taken together, staff finds the observations and measurements provide a coherent supporting basis for the conceptual site model. Where observations and measurements may not be consistent with site characteristics and the implemented site conceptual flow model, specifically, the sorption coefficients, OPPD committed in its response to TE2-11 (ML23236A478) to provide a report to reconcile the discrepancy. The report will include the data on sorption characteristics from laboratory tests performed on samples from the site and the licensee's reconciliation of laboratory results with sorption coefficient inputs in the LTP. For the reasons discussed above, the NRC staff found that the properties of the site that support the groundwater dose assessment input parameters adequately represent site conditions, or that OPPD performed sensitivity analyses or used or will use conservative values to address any significant uncertainty.

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[16] Staff notes that NUREG/CR-5512 Volume 3 was not finalized and its status remains as "draft for comment." However, staff considers the irrigation rates from the volume to be endorsed by NRC because they are used as default values in the DandD code. The NRC endorses the DandD code for use in decommissioning.

The FCS LTP identified historical radiological spills, leaks, and releases with the potential to have impacted groundwater quality. While OPPD has reported detectable concentrations of radionuclides in the groundwater near the FCS footprint and made revisions to the LTP to provide information for the FSS. Pending confirmation by FSS measurements that no releases were identified or occurred during remaining decommissioning activities, Concentrations for radiological constituents at the monitoring well locations appear to have remained below EPA maximum contaminant level for drinking water. The NRC staff have identified no imminent threats to human health or the environment due to radiological constituents in the groundwater, and OPPD will perform ongoing additional assessments, i.e., groundwater monitoring, during the remainder of decommissioning.

Staff has reviewed and concluded that the licensee provided adequate description and support for the site characteristics that are important for the development of a site conceptual model and for parameter inputs used in the dose assessment models. Staff finds that the licensee's geological and hydrogeological site characterization is sufficiently detailed to fulfill the site characterization requirement of 10 CFR 50.82(a)(9)(ii)(A). For the reasons described above, staff have reasonable assurance information provided on geology and hydrology is appropriate for supporting inputs to dose models that demonstrate compliance with the unrestricted release criteria specified in 10 CFR 20.1402.

6.13 Compliance with Radiological Criteria for License Termination Conclusions

The NRC staff reviewed the information provided in the LTP pertaining to the licensee's assessment of the potential doses resulting from exposure to residual radioactivity remaining at the end of the decommissioning process. This review was conducted using section 2.6 of NUREG-1700, which refers to section 5.2 and appendix I of NUREG-1757, Volume 2, Revision 2, to conclude that the LTP is in compliance with the unrestricted release criteria specified in 10 CFR 20.1402.

To summarize, the NRC staff has reasonable assurance of the following:

- The licensee has adequately characterized and applied its source term.
- The licensee has analyzed the appropriate dose scenario(s), and the exposure group(s) adequately represents a critical group.
- The mathematical method and parameters used are appropriate for the dose scenario(s), and parameter uncertainty has been adequately addressed.
- The peak annual dose to the average member of the critical group for the appropriate exposure scenario(s) was used to calculate the BcDCGLs.
- The licensee has committed to using radionuclide-specific DCGLs, and will ensure that the total dose from all radionuclides and all sources (soil, backfilled basements, buried piping, and remaining aboveground structures) will meet the requirements of Subpart E of 10 CFR Part 20 using the sum of fractions approach and the compliance dose equations in section 6.1 of this SER.

7 SITE SPECIFIC COST ESTIMATE

In accordance with the requirements of 10 CFR 50.82(a)(9)(ii)(F) and consistent with the guidance of RG 1.179, Chapter 7, "Update of the Site-Specific Decommissioning Costs," of the FCS LTP, Revision 1, provides an updated site-specific estimate of remaining decommissioning costs to complete the dismantlement and decontamination activities at the FCS site. This portion of the LTP estimates the decommissioning costs remaining at the time of LTP submittal and compares the estimated remaining costs with the present funds set aside for decommissioning. If there is a deficit in present funding, then the LTP will indicate the means for ensuring adequate funds to complete the decommissioning activities.

7.1 Financial Requirements and Cost Estimate Criteria

10 CFR 50.82(a)(9) states in part: "All power reactor licensees must submit an application for termination of license. The application for termination of license must be accompanied or preceded by a license termination plan to be submitted for NRC approval."

According to 10 CFR 50.82(a)(9)(ii)(F), for NRC to evaluate and provide approval of the LTP, the submittal should include an updated site-specific estimate of remaining decommissioning costs.

In accordance with 10 CFR 50.82(a)(9)(ii)(F), chapter 7 of the LTP provides an updated estimate of the remaining decommissioning costs for releasing the FCS site for unrestricted use. This chapter also compares the estimated remaining cost with the funds currently available in the decommissioning trust fund.

7.2 Evaluation of the Updated Site-Specific Decommissioning Cost Estimate

As required by 10 CFR 50.82(a)(9)(ii)(F), OPPD estimated the remaining decommissioning costs associated with the termination of the FCS license to be $591.0 million (2020 dollars). OPPD based its cost estimate on the fundamental technical approach in AIF/NESP-036, "Guidelines for Producing Commercial Nuclear Power Plant Decommissioning Cost Estimates." The NRC endorsed AIF/NESP-036 in NUREG-1713, "Standard Review Plan for Decommissioning Cost Estimates for Nuclear Power Reactors" (ML043510113), meaning it is the NRC's preferred method for how licensees meet the relevant regulatory requirements. According to the LTP, OPPD has used a cost model that includes elements for estimating distributed and undistributed costs. Distributed costs are activity-specific and include planning and preparation costs as well as costs for decontamination, packaging, disposal, and removal of major components and systems. Costs for the segmentation, packaging, and disposal of the reactor internals are distributed costs. Undistributed costs, sometimes referred to as collateral costs, are time-dependent costs such as utility (Licensee) and decommissioning general contractor staff, property taxes, insurance, regulatory fees and permits, energy costs, and security staff.

Prior to starting the detailed review of the cost estimate, the NRC reviewed the estimate to confirm the supporting systems/structures necessary to support the safe operation had been identified in the estimate. The NRC determined that the cost estimate is based on a reasonable estimate of the cost to decommission the remaining supporting systems and structures, included all the major equipment necessary to support operation was included.

In the LTP, the licensee has divided the estimated remaining decommissioning costs (2020 dollars) into the following time periods: Period One: Spent Fuel Pool to Pad; Period Two: Decommissioning and License Termination; and Period Three: Spent Nuclear Fuel and Greater-than-Class C dry storage and disposition. The estimated total remaining cost of decommissioning based on the above factors, as of December 2020, is $591.0 million. This estimate includes overall average contingency of 10.2 percent across the major activities. The staff reviewed the contingency factors and the work difficulty factors used in the OPPD cost estimate and found them to be reasonable because the estimate is consistent with the NRC-endorsed, industry technical standards of decommissioning cost estimating under AIF/NESP-036.

7.3 Evaluation of the Decommissioning Funding Plan

According to the LTP, as required by regulation, OPPD maintains an external trust fund for the decommissioning of FCS. The LTP affirms that the market value of the trust fund account as of December 31, 2020, is $337.8 million. In addition, OPPD has accumulated funds in a separate decommissioning account based on the difference between the site-specific study's estimated cost to fully decommission FCS and the NRC's regulated formula-based cost to decommission the radiologically contaminated portions of FCS. The market value of that decommissioning fund account was $204.3 million as of December 31, 2020. So, the total available for License termination as of the end of 2020 is $542.1 million. Staff notes that the decommissioning funding amount available is 92% of the estimated cost to fully decommission the plant.

In addition to the requirement to provide an updated site-specific estimate of remaining decommissioning costs for the evaluation of the LTP, OPPD is also required to provide annual decommissioning funding status reports for FCS. Pursuant to 10 CFR 50.82(a)(8)(V), the licensee is required to provide specific information related to decommissioning costs, expenditures, and funds. Accordingly, on March 30, 2022, OPPD submitted 2021 FCS Decommissioning Funding Status (DFS) Report (ML22090A078). The subsequent NRC analysis of the annual DFS report was based on a Decommissioning Trust Fund balance for radiological decommissioning of $494.4 million as of December 31, 2021. The licensee applied a real rate of return of less than 2.0 percent, allowed by regulation, to its analysis through the expected license termination year of 2059 and credited additional OPPD-authorized collections of about $110 million per year for years 2022 – 2024, totaling $323 million in future collections, resulting in a surplus of funds over the estimated $591.0 million needed to complete decommissioning.

The NRC staff finds the site-specific cost estimate for remaining radiological decommissioning costs for FCS is reasonable, and that the Decommissioning Trust Fund balance, as of December 31, 2021, will be sufficient to fund the remaining radiological decommissioning expenses.

7.4 Site Specific Cost Estimate Conclusions

The NRC finds that decommissioning cost estimate and decommissioning funding plan associated with OPPD's LTP for FCS are adequate and provide sufficient details associated with the funding mechanisms for decommissioning activities to be accomplished. The NRC, therefore, concludes that the licensee's LTP for OPPD Unit 1 complies with 10 CFR 50.82(a)(9)(ii)(F).

8 ENVIRONMENTAL CONSIDERATIONS

In accordance with 10 CFR 50.82(a)(9)(ii)(G), requiring a supplement to the environmental report (ER) describing any new information or significant environmental change associated with the licensee's proposed termination activities, OPPD submitted an ER supplement as chapter 8 of the LTP. Pursuant to 10 CFR 51.21 (stating criteria for and identification of licensing and regulatory actions requiring environmental assessments), 10 CFR 51.32 (addressing a finding of no significant impact), and 10 CFR 51.35 (providing the requirement to publish a finding of no significant impact, and limiting Commission actions pre-publication of the finding of no significant impact), a notice of the issuance of the NRC's environmental assessment and finding of no significant impact were published in the *Federal Register* on January 2, 2024 (89 *FR* 105). The environmental assessment and finding of no significant impact were published on December 31, 2023 (ML23333A049). Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

9 PARTIAL SITE RELEASE CONSIDERATIONS

In accordance with 10 CFR 50.82(a)(9)(ii)(H) the LTP shall identify the parts of the facility or site that were released for use before approval of the LTP. On June 29, 2018 (ML18215A187), OPPD requested approval from the NRC to remove approximately 120 acres of land on the northwest portion of the Owner-Controlled Area (OCA) from the Part 50 License. On November 12, 2018 (ML18316A036), OPPD submitted a request for partial site release of the 475-acre property in Iowa that was subject to a perpetual easement. OPPD requested the release to these areas because they had been classified by the licensee as radiologically non-impacted.

NRC published a *Federal Register* Notice on November 26, 2018 (83 *FR* 60508) requesting comment on the OPPD partial site release requests and also held a public meeting in the vicinity of FCS on November 28, 2018, to obtain public comments associated with the partial site releases (see public meeting summary dated March 13, 2019 (ML19071A030)). No comments were received that were in scope of the request from OPPD.

In accordance with 10 CFR 50.83, NRC approved the releases by letter dated April 10, 2019 (19074A301). Before approval of the releases, NRC verified that the areas to be released had not been radiologically impacted by licensed site activities, as described in the report entitled: "Confirmatory Survey Summary and Results for the Non-Impacted Land Areas Associated with the Fort Calhoun Nuclear Generating Station Blair, Nebraska" that was prepared for the NRC by ORISE, dated February 27, 2019 (ML19065A256). The ORISE confirmatory survey, performed November 27-29, 2018, included surface gamma scanning and collection of randomly selected volumetric soil samples. It identified no elevated direct radiation during surface gamma scanning. It attributed radionuclides that it identified in soil samples to natural sources and not site operations. ORISE did not identify anomalies that would prevent the classification subject areas as non-impacted.

Section 1.3.2 of the LTP identifies the parts of the site that were released from the license prior to approval of the LTP, thereby satisfying the requirement in 10 CFR 50.82(a)(9)(ii)(H).

10 EPA MOU

By letter dated June 26, 2023 (ML23082A220), NRC notified the U.S. Environmental Protection Agency (EPA), in accordance with the Memorandum of Understanding (MOU) on "Consultation and Finality on Decommissioning and Decontamination of Contaminated Sites" dated October 9, 2002 (ML022830208) between NRC and EPA. The MOU contains criteria related to the DCGL soil concentrations proposed by licensees if those proposed concentrations in an LTP or decommissioning plan exceeds any of three trigger criteria contained in the MOU. For sites that trigger the criteria, the NRC will consult with the EPA at two points in the decommissioning process: (1) prior to NRC approval of the LTP or decommissioning plan, which the NRC terms Level 1 consultation; and (2) following completion of the Final Status Survey (FSS), which the NRC terms Level 2 consultation. The NRC letter to EPA was a Level 1 consultation because the licensee's proposed DCGLs for certain radionuclides at this site exceed the soil concentration values in table 1 of the MOU.

By letter dated September 28, 2023 (ML23276A004), EPA responded to the NRC Level 1 consultation letter by providing its views and comments on the licensee's proposed DCGLs and their understanding that the remediation activities associated with NRC's decommissioning process are likely to significantly decrease the levels of those radionuclides that are present to residual levels below the DCGLs.

If FSS measurements show that the remaining radionuclide concentrations are below the values set forth in table 1 of the MOU as well as the final approved DCGL values, then the NRC will proceed to terminate the FCS license (except for the ISFSI) and the site will be released for unrestricted use. The NRC will inform the EPA of such findings. If the FSS measurements show that the remaining radionuclide concentrations are above the values set forth in table 1 of the MOU, then the NRC will engage in Level 2 consultation with the EPA to identify and resolve any remaining issues.

11 STATE CONSULTATION

In accordance with the Commission's regulations, the Nebraska State official, Becki Harisis, Manager Public Health - Office of Radiological Health, Nebraska Department of Health & Human Services, was notified of the proposed issuance of the amendment on January 22, 2024. The State official responded by e-mail dated January 23, 2024 (ML24025A179) and did not have any comments or objections to issuing the amendment.

12 CONCLUSIONS

The NRC has concluded, based on the considerations discussed above, that there is reasonable assurance that the remainder of the decommissioning activities at FCS, as described in the LTP (1) will be performed in accordance with the regulations in 10 CFR Part 50; (2) will not be inimical to the common defense and security or to the health and safety of the public; and (3) will not have a significant effect on the quality of the environment.