SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

RELATED TO AMENDMENT NO. 303

TO FACILITY OPERATING LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT 1

DOCKET NO. 50-285

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1 INTRODUCTION

The Omaha Public Power District's (OPPD), Fort Calhoun Station, Unit 1 (FCS), is a permanently shutdown nuclear power reactor in active decommissioning. The goal of FCS decommissioning is to complete all activities necessary to meet the NRC's requirements for license termination in Section 20.1402, "Radiological criteria for unrestricted use," of Title 10 of the *Code of Federal Regulations* (10 CFR). Meeting these requirements would allow for the termination of the license and release the FCS site for unrestricted use, except for the footprint of the Independent Spent Fuel Storage Installation (ISFSI) containing the spent nuclear fuel. The decommissioning activities and timeline are described in the FCS Post Shutdown Decommissioning Activities Report (PSDAR) dated December 16, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19351E355). In accordance with Paragraph (a)(11) of 10 CFR 50.82, "Termination of license," the U.S. Nuclear Regulatory Commission (NRC) shall terminate a power reactor license if it determines that: (i) the remaining dismantlement has been performed in accordance with the approved license termination plan, and (ii) the final radiation survey and associated documentation demonstrate that the facility and site have met the criteria for license termination .

2 BACKGROUND

Revision 0 of the FCS License Termination Plan (LTP) was submitted to the NRC under the requirements of 10 CFR 50.82(a)(9) on August 3, 2021 (ML21271A178). After review of Revision 0 by the NRC, but before approval, Revision 1 of the FCS LTP (LTP Rev. 1) was submitted on December 6, 2023 (ML23346A152). Under the provisions of 10 CFR 50.82(a)(1), FCS LTP Rev. 1 was approved by the NRC on January 31, 2024, as documented in the NRC staff's Safety Evaluation Report (SER) (ML24019A168).

2.1 License Amendment Request

OPPD submitted an application titled "License Amendment Request (LAR) To Revise License Termination Plan" to the NRC on June 18, 2024 (ML24177A132) that included Revision 2 of the FCS LTP (LTP Rev. 2). The revision addressed changes needed for the grouting of excavated trenches in the Auxiliary Building. OPPD provided supplemental information for LTP Rev. 2 on October 2, 2024 (ML24276A217) and January 16, 2025 (ML25016A212). The NRC initiated a regulatory audit of FCS LTP Rev. 2 on January 14, 2025 (ML25023A155). The NRC closed the audit and issued a summary and Request for Additional Information (RAI) on June 18, 2025 (ML25154A358). Following the audit, OPPD submitted a response to the audit on July 28, 2025 (ML25211A290), and a "Supplement to the License Amendment Request (LAR) To Revise License Termination Plan (LTP)" to the NRC on July 31, 2025 (ML25218A293), that expanded the changes to LTP Rev. 2 to include additional grouted areas in subsurface structures and revisions to additional areas to address issues identified during the audit.

In the initial request, OPPD submitted the license amendment request to approve Revision 2 of the FCS LTP in accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," to revise the dose calculations for the Auxiliary Building basement and the requirements for remediation and survey methodologies approved in LTP

Rev. 1. Specifically, OPPD requested the amendment to change the remediation strategy for the embedded pipe in the Auxiliary Building basement floors from “decontaminate and leave in place” to removal of most of the embedded piping. The licensee removed a majority of the activity along with the piping, but significant contamination remained in the concrete trenches created by the pipe removal. The trench remediation also damaged the concrete foundation. This damage resulted in foundation fractures that allowed groundwater intrusion. Therefore, OPPD made a decision to grout the trenches to eliminate additional groundwater in-leakage. Prior to grouting the trenches, OPPD needed to complete a Final Status Survey (FSS) to demonstrate that the Derived Concentration Guideline Limits (DCGLs) approved in LTP Rev. 1 were satisfied. However, the trench radionuclide mixture and concentrations, and the trench geometries, created challenges in meeting the survey requirements for FSS. To expedite the remediation and FSS of the trenches, OPPD decided to take credit for the grout as a mitigation measure in the dose calculations for determining alternate DCGLs to meet the dose criteria for license termination in 10 CFR Part 20, “Standards for Protection Against Radiation,” Subpart E, “Radiological Criteria for License Termination.”

In the supplement to the LAR and revision to FCS LTP Rev. 2, OPPD applied the grouting approach to two additional areas: the tendon stressing gallery and a portion of the concrete wall at the 973 foot (973-ft) (296.6 meter(m)) elevation, both in the Containment Building. OPPD implemented the FSS and dose modeling approach for grouting to these areas in the Containment Building using different mixture fractions compared to those in the Auxiliary Building trenches based on the continuing characterization data.

Although the need for NRC approval applies only to the methodology changes, minor changes in other chapters of the FCS LTP were also made in relation to the proposed changes to alleviate confusion and facilitate NRC review. Each of these changes is evaluated, as necessary, as part of this SER. In the remainder of the SER, the NRC staff refers to the supplement dated July 31, 2025, as simply FCS LTP Rev. 2.

2.2 Regulatory Requirements

The NRC staff accepted the proposed LAR and FCS LTP Rev. 2 for review by letter dated November 18, 2024 (ML24318C253), and published a notice of consideration of the proposed LAR and a finding of no significant hazard consideration determination in the *Federal Register* on December 23, 2024 (89 FR 104568). 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.

3 TECHNICAL EVALUATION

OPPD submitted a LAR that incorporated the use of grout as an engineered barrier for selected subsurface structures at the site. The LAR, as supplemented, describes the substantial changes to the site remediation plan, final status survey plan, and dose models for the three grouted areas that did not meet the FCS LTP Rev. 1 DCGLs or were not considered at the time of the LTP Rev. 1 approval. The licensee identified the three grouted areas as (i) the east trenches in floor slabs of the Auxiliary Building, (ii) a portion of the west wall of the Containment Building, and (iii) the tendon stressing gallery in the Containment Building. These three grouted areas are described in Sections 6.13, 6.14, and 6.15 of the LTP Rev.2, as well as in licensee documents: FC-25-001, “Radionuclide Fractions and Surrogate Ratios for the FCS Auxiliary Building 971' Elevation,” Revision 0 (ML26027A173), and FC-25-002, “Radionuclide Fractions for the FCS Containment Building,” Revision 0 (ML26027A166). Additional details were also provided in

FC-24-004, “DCGLs, Remediation Strategy and Survey Methodology for the Auxiliary Building East Trenches,” Revision 1 (ML26027A171). Discussions of grouted areas in this SER only refer to the three areas mentioned above. The NRC staff notes that conclusions associated with other grouted areas (e.g., other embedded pipes or the discharge structure) have not changed from the SER approving Revision 1 of the FCS LTP.

The NRC staff’s technical evaluation of Revision 2 of the FCS LTP covers four topical areas: (i) as low as reasonably achievable (ALARA) analyses in support of changes to the FSS plan, (ii) updated final status survey plans, (iii) revised scenarios and dose models for the Auxiliary and Containment Buildings, and (iv) application of a diffusion release model for grouted areas. The NRC staff’s evaluation of these four topics is provided in Sections 3.1 through 3.4 of this SER, respectively.

3.1 Site Remediation Plan and ALARA Analysis

The licensee made several changes to various parameters used in their ALARA analysis. This included updated values to reflect more recent years, inflation adjustments, and new DCGLs and doses for individual scenarios. The updates culminated in a revised ALARA analysis in the LTP Rev. 2, Section 4.4.1.5, “Residual Radioactivity in Soils that are ALARA,” and the LTP Rev. 2 Table 4-5, “ALARA Analysis for the Auxiliary Building Basement.” These updated evaluations did not result in any changes to the conclusions of the ALARA analysis.

In addition, Chapter 4, “Site Remediation Plan,” of the LTP Rev. 2 was updated by the licensee to include Section 4.5, “ALARA Considerations for Grout as a Mitigation Measure,” which stated that they intend to use grout as a mitigation measure in limited circumstances, particularly in concrete trenches. This is consistent with NUREG-1757, Volume 2, “Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria – Final Report” (ML22194A859), Section 3.5, “Use of Engineered Barriers,” and Appendix P, “Framework for Use of Engineered Barriers at Decommissioning Sites,” which provide guidance to licensees considering the use of engineered barriers, such as stabilizing cementitious materials, to demonstrate compliance with the radiological criteria for license termination. However, as NUREG-1757 states for licensees pursuing unrestricted use of their site, residual radioactivity must be reduced to levels that are ALARA before any reliance on engineered barriers to meet the License Termination Rule criteria. The licensee stated that residual radioactivity in the trenches will be reduced to levels that are ALARA without factoring the reduction in release due to grouting, which is considered an engineered barrier. Also, the licensee updated their site-specific calculations, which evaluated concrete scabbling or shaving remediation activity for the Auxiliary Building basement floor in the LTP Rev. 1, to include updated Basement DCGLs for the Auxiliary Building.

3.1.1 <u>NRC Evaluation of Site Remediation Plan and ALARA Analysis</u>

As stated in the LTP Rev. 2, Section 4.4.2, “ALARA Analysis for Remediation of Basement Structures,” the ALARA assessment of the Auxiliary Building Basement floor at FCS bounds ALARA assessments for other buildings that use the same remediation method (scabbling). The licensee’s ALARA analysis for remediation of soils and basement structures was already approved as part of LTP Rev. 1 and as noted in revised LTP Rev. 2 Section 4.5; however, the licensee is still required to reduce residual radioactivity to levels that are ALARA before reliance on engineered barriers such as grout. The NRC staff did not re-evaluate the licensee’s method for analysis but only verified updates to parameter values and calculations.

The NRC staff verified the updated licensee parameters and calculations from the LTP Rev. 2, Section 4.4, "ALARA Evaluation," used as input into Section 4.4.1.5 and Table 4-5. In addition, the NRC staff verified the licensee's calculations in the ALARA analysis presented in Table 4-5 using the licensee's updated parameters and DCGLs and found all derivations are greater than unity, which demonstrates the criteria being applied are considered ALARA.

Additionally, during the NRC staff's review of the licensee's ALARA calculations utilizing equations from Appendix N, "ALARA Analyses," of NUREG-1757, Volume 2, Revision 2, the staff identified two errors. The first error was in the calculation of $Cost_{PDose,}$ which is included in the total cost ($Cost_T$). The second error was in the value of averted dose (V_{AD}) in Table 4-5. Both parameters are ultimately used to determine the $Conc_{ALARA}/DCGL_W$ ratio, which if greater than 1, validates that further remediation beyond that required to demonstrate compliance with the 0.25 milliSievert per year (25 millirem per year) dose criterion is not justified. In both cases, the errors resulted in a slightly lower $Cost_{ALARA}/DCGL_W$ ratio; however, the corrected calculation of $Conc_{ALARA}/DCGL_W$ by the NRC produced the same outcome (i.e., $Conc_{ALARA}/DCGL_W > 1$), thus confirming the licensee's analysis.

As a result of this analysis, the NRC staff was able to demonstrate that the FCS LTP Rev. 2 meets the acceptance criteria as delineated in NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," Revision 2 (ML18116A124) (SRP), Section 2.4, "Remediation Plans," which is one way to meet the regulatory requirements. Based on this review, the NRC staff 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 "plans for site remediation"). In particular, the NRC staff find that the licensee's ALARA analysis is acceptable for all three grouted areas because the extent and type of remediation assumed in the ALARA analysis bounds that applied to those areas covered by the LTP Rev. 2 (i.e., trenches in the Auxiliary Building and both the 973-ft (296.6-m) elevation of the west wall and tendon stressing gallery of the Containment Building).

3.2 Final Status Survey Plan

The licensee revised Chapter 5, "Final Status Survey Plan" in the LTP Rev. 2 to incorporate updated DCGLs for (i) the Auxiliary Building (AB) walls, floors, and grouted trenches and (ii) the Containment Building (CB) tendon stressing gallery and a portion of the concrete wall on the 973-ft (296.6-m) elevation. The licensee also updated mixture fractions in these areas which are used to develop insignificant contributors (ICs), radionuclides of concern (ROCs), and surrogate ratios related to radiological dose calculations for areas in the AB and CB. In addition, the licensee provided additional information on FSS survey methodologies to allow for traditional methods (hand scanning and volumetric samples for compliance) and provisions for using area factors and $DCGL_{EMC}$ in basement structures and embedded piping.

The licensee also made changes to language regarding investigations and remediation and how these would be conducted, but did not include this in Attachment 1, Section 2.0, "Detailed Description," which contains an overview of changes made in the LTP Rev. 2. According to the LTP Rev. 2, post-demolition remedial action support survey (RASS) investigations can now be conducted using an In Situ Object Counting System (ISOCS) with a reduced field of view (FOV) or hand-held gamma scanning, and, according to FC-24-004, FSS investigations in the Auxiliary Building East Trenches will be performed with a Ludlum Model 44-10 detector. FCS LTP Rev. 2 further stated that basement and embedded pipe areas above $DCGL_{EMC}$ will now be remediated, rather than using the previous BcDCGL level. Further discussion and evaluation of these changes can be found in the subsections below.

3.2.1 Radionuclides of Concern, Mixture Fractions, and Insignificant Contributors

FCS LTP Rev. 2., Section 5.2.5, “Radionuclides of Concern and Mixture Fractions,” and the following subsections include a new evaluation of the anticipated fractional makeup, relative doses and the selection of ROCs for various areas in the AB and CB. This includes 7 new mixture fractions developed for the following areas:

- AB East Trenches
- AB 971’ Ungrouted Structures (SU 2102)
- AB 971’ Grouted Structures (SU 2103 and SU 2104)
- CB Liner (SU 1000)
- CB 973-ft Grouted Structures (SU 1001)
- CB Stressing Gallery (SU 1002)
- CB Safety Injection Recirculation Line (SIRL) (SU 9009)

The licensee included a detailed discussion of these mixture fractions in technical support documents (TSDs) FC-24-004, FC-25-001, and FC-25-002. The development of radionuclide fractions, relative doses, and selection of ROCs followed the same method that was approved in TSD FC-21-043, “Radionuclides of Concern in Support of the Fort Calhoun License Termination Plan” (ML26027A177), during the review of the LTP Rev. 1. In each case, the licensee collected volumetric samples from applicable areas and sent them to GEL Laboratories, LLC for analysis of the full suite of radionuclides previously established in FC-21-043. In most cases, the licensee developed mixture fractions using the 75^{th} percentile of isotope fractions as related to Cesium-137 (Cs-137) (i.e., the ratio of any particular radionuclide concentration to that of Cs-137 concentration in the same sample). The licensee used this particular method in previously reviewed and approved technical support documents as a common and conservative method to ensure that the final selection of radionuclide mixtures accounts for variability and represents conservative estimates. However, for two survey units (SU 1000 and SU 1001), the licensee chose to develop mixture fractions using the normalized 75^{th} percentile of the population of samples. According to the licensee, OPPD chose this option due to either low Cs-137 concentrations resulting from neutron activation as a contamination source, or sampling and analysis considerations as further described in FC-25-002.

By dividing the chosen mixture fraction by the applicable DCGLs for each area, the licensee then established a relative dose contribution from each initial ROC. OPPD deselected ICs in accordance with NUREG-1757, Volume 2, Section 3.3, “Insignificant Radionuclides and Exposure Pathways,” and selected the remaining radionuclides as dose significant ROCs. The dose significant ROCs and renormalized mixture fractions developed in the technical support documents for all new areas are included in the LTP Rev. 2 Table 5-3, “AB Dose Significant Radionuclides and Renormalized Mixture Fractions,” Table 5-4, “CB Dose Significant Radionuclides and Renormalized Mixture Fractions,” and Table 5-5, “TB, TB Embedded Piping, RWBP, Soil, Buried Piping, and Basement Fill Dose Significant Radionuclides and Renormalized Mixture Fractions.” The IC dose from the deselected radionuclides in all cases is less than 5 percent, except for SU 1001 (CB 973-ft Elevation Grouted West Wall Concrete), where the calculated IC dose is 7.58 percent. This matches the IC dose adjustment for all DCGLs in the LTP Rev. 2 Tables 5-7 through 5-17, which was 5 percent in all cases except for the CB 973-ft Elevation Grouted West Wall Concrete, where the IC dose adjustment was 10 percent, as outlined in the supplemental information provided by OPPD (ML25247A055).

The NRC staff notes that according to FC-24-004, the licensee also conducted a RASS for the AB East Trenches and collected 12 concrete core samples for analysis to confirm surrogate ratios, but not ICs. During the NRC audit (ML25154A358), the licensee noted that a verification of IC dose for the AB East Trenches was not completed because OPPD conducted characterization surveys so close in time to the RASS, and as such did not have a valid concern that the dose contributions of ICs could be significant following any remediation.

With regard to the potential for future development of additional mixture fractions, the licensee states in the LTP Rev. 2, Section 5.2.5.1, "Radionuclides of Concern and Mixture Fractions for the Auxiliary Building," that for AB 989-ft (301.5-m) elevation grouted basement structures other than the AB East Trenches, the particular ROCs for a given structure will be determined on a case-by-case basis, given the location of the survey unit and the results of any characterization, radiological assessment (RA), or RASS. Justifications for the selected ROCs will be provided in the release record for each AB 989-ft (301.5-m) elevation grouted basement structure SU.

3.2.1.1 NRC Evaluation of ROCs, Mixture Fractions, and ICs

The NRC staff evaluated the licensee's proposed ROCs, mixture fractions, and ICs in accordance with the regulatory guidance and acceptance criteria contained in Section 2.5, "Final Radiation Survey Plan," of NUREG-1700 and Section 3.3, "Insignificant Radionuclides and Exposure Pathways," of NUREG-1757, Volume 2, Revision 2. The NRC staff found the amount and selection of samples used to establish ROCs, as well as the licensee's use of decay corrected activity and the 75th percentile values to determine mixture fractions and ICs, reasonable in the LTP Rev. 2 and supporting documentation. In one case, for the CB SIRL (SU 9009), OPPD took only 4 metal "coupon" samples to establish a mixture fraction and develop ROCs; however, the NRC staff found the licensee's justification, which cited the difficulty of obtaining samples due to the pipe's configuration within walls, to be adequate because the staff determined that 4 coupons was enough to characterize the variability sufficiently to provide reasonable assurance. The NRC staff also verified a subset of the newly developed mixture fractions, relative dose fractions, and IC evaluations found in FC-24-004, FC-25-001, and FC-25-002. While the NRC staff identified small inconsistencies in several of the tables, the staff reproduced the OPPD's mixture fractions, relative dose and dose significant radionuclides within rounding. With regard to IC dose contribution, the NRC staff-calculated values varied minimally from licensee calculated values but ultimately are still less than the IC dose correction that the licensee applied to the corresponding DCGLs for these areas.

The NRC staff notes that this review only covers the 7 areas of the AB and CB described above. Justification for new ROCs, mixture fractions, and ICs in the AB 989-ft (301.5-m) elevation grouted basement structures other than the AB East Trenches (as described in the LTP Rev. 2, Section 5.2.5.1) will need to be included in a future FSS Report for review. If newly developed mixture fractions or other changes result in a change to any of the DCGLs, an amendment to the LTP will need to be submitted for NRC review and approval.

Based on this evaluation, the NRC staff finds the revised mixture fractions, ROCs, and ICs identified in the LTP Rev. 2 and referenced in the technical support documents to be acceptable because they are reasonable and appropriate for the FCS site, are consistent with guidance found in the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)

(NUREG-1575, Revision 1) and NUREG-1757, and demonstrate compliance, in part, with the final radiation survey requirements of 10 CFR 50.82(a)(9)(ii)(D).

3.2.2 Surrogate Radionuclides

FCS LTP Rev .2, Section 5.2.6.2.2, “Surrogate Radionuclides for Auxiliary Building Grouted Basement Structures,” and Section 5.2.6.2.3, “Surrogate Radionuclides for the Auxiliary Building 971’ Elevation,” discuss the surrogate ratios developed for areas using the new ROCs established in FC-24-004 and FC-25-001.

In the AB East Trenches, where the ROCs are Cobalt-60 (Co-60), Cs-137, and tritium (H-3), the licensee selected Cs-137 as a suitable surrogate for H-3 consistent with the surrogate method outlined in MARSSIM. OPPD sent 7 continuing characterization samples to GEL for full suite analysis. All 7 results showed Cs-137 concentrations greater than the minimum detectable concentration (MDC) and 2 out of 7 showed H-3 concentrations greater than MDC. In the instances where GEL did not detect H-3 at levels above MDC, the licensee assigned the decay corrected MDC as the H-3 activity for the ratio calculation shown in FC-24-004, Table 25, “Continuing Characterization H-3/Cs-137 Ratio.” The licensee selected the 95th percentile values as the final surrogate ratio to account for the variability and level of uncertainty in the data. This, along with the MDC substituted in cases where H-3 concentrations were less than MDC, provides reasonable conservatism during final status surveys.

The licensee also collected 12 additional samples during RASS in the AB East Trenches and compared the H-3/Cs-137 ratio to the ratio from continuing characterization. Of these 12 samples, one had a surrogate ratio higher than that found during continuing characterization, and as required by the LTP Rev. 2, Section 5.2.5, OPPD selected this value as the new surrogate ratio value for the sample unit. The final ratio for the AB East Trenches is summarized in the LTP Rev. 2,Table 5-24, “Surrogate Ratio for AB East Trenches”. The LTP Rev. 2 states that ratios from this table will be used to infer hard-to-detect (HTD) ROCs unless actual measured concentrations are used instead of inference.

The licensee conducted a similar evaluation to establish surrogate ratios for the Auxiliary Building 971-ft Elevation Ungrouted Walls (SU 2102) documented in FC-25-001. The ROCs are Co-60, Nickel-63 (Ni-63), Strontium-90 (Sr-90), and Cs-137. OPPD selected Cs-137 as the most appropriate surrogate radionuclide for both Ni-63 and Sr-90. The licensee based the selection for Cs-137 as the surrogate for Ni-63 on a review of ISOCS and offsite laboratory results. OPPD sent 8 volumetric samples to GEL for full suite analysis. Four of the Ni-63, 2 of the Sr-90 and all of the Cs-137 results showed concentrations greater than MDC. In the instances where GEL did not detect Ni-63 and Sr-90 at levels above MDC, the licensee selected the decay corrected MDC as shown in FC-25-001, Table 22, “Ni-63 and Cs-137 Decayed Reported Concentrations, MDCs and Ratios,” and Table 23, “Sr-90 and Cs-137 Decayed Reported Concentrations, MDCs and Ratios.”

The NRC staff notes that an error shows up in both tables where the Cs-137 MDC is listed as the value that will be used even though all reported results showed concentrations greater than MDC; however, the final calculated ratios use the correct Cs-137 values. Similar to the AB East Trenches, the licensee selected the 95th percentile values as the final surrogate ratio. During FSS of SU 2102, OPPD collected 13 concrete core samples and sent those to GEL for analysis that included Ni-63 and Sr-90. Three samples had positive results for both Cs-137 and Ni-63, but because all 3 calculated surrogate ratios were less than the ratio previously calculated, OPPD used the previously calculated ratio. One sample had positive results for both Cs-137 and Sr-90. The calculated surrogate ratio for this sample slightly exceeded the ratio previously calculated, so OPPD selected the new value for the calculations used in the final data assessment in the Survey Unit Release Record. This approach is conservative in both cases.

The licensee states that for grouted basement structures in the AB 989-ft (301.5-m) elevation other than the AB East Trenches, the particular ROCs and surrogate ratios for a given structure will be determined on a case-by-case basis, given the location of the survey unit and the results of any characterization, RA, or RASS. Justification for the ROCs will be included in the survey unit release records included with the FSS Report. The LTP Rev. 2 states that ratios will be used to infer HTD ROCs unless actual measured concentrations are used.

3.2.2.1 NRC Evaluation of Surrogate Radionuclides

The NRC staff notes this review only covers the AB East Trenches and 971-ft Elevation Ungrouted Walls. Discussion and justification for surrogate ratios in other grouted basement structures will need to be included in a future FSS Report for review.

The NRC staff evaluated the licensee's use of surrogates in the LTP Rev. 2 in accordance with the regulatory guidance and acceptance criteria contained in Section 2.5 of NUREG-1700 and MARSSIM Section 4.3.2, "DCGLs and the Use of Surrogate Measurements." The NRC staff found the licensee's use of the 95th percentile values to establish initial surrogate ratios to be reasonable in the LTP Rev. 2 and supporting documentation. Furthermore, the licensee's use of RASS or FSS samples to identify non-conservative variations in the surrogate ratios is acceptable because they are reasonable and appropriate for the FCS site, are consistent with the applicable guidance, and demonstrate compliance, in part, with 10 CFR 50.82(a)(9)(ii)(D).

3.2.3 <u>Area Factors</u>

FCS LTP Rev. 2, Section 5.2.6.5, "Area Factors," includes additional language on the use of area factors for elevated areas of contamination located on a basement wall or floor (including grouted basement structures) or in an embedded pipe and defines how area factors in these areas will be established.

During the NRC audit discussions (ML25154A358), the licensee provided additional details on how the "simple calculation" for area factors, which is based solely on the size of a survey unit without consideration for the specific radionuclides, exposure scenarios and applicable modeling parameters, can be applied to both soils associated with the resident farmer scenario and concrete/grout associated with an industrial use scenario. OPPD also noted that they included this "simple calculation" in the previously approved LTP Rev. 1 and that it is the standard area factor equation used to determine the activity in an elevated area that equates to the total activity in the survey unit.

Because this approach has been previously reviewed and approved, and the revised language in the LTP Rev. 2 is administrative in nature, further NRC staff evaluation of the LTP Rev. 2, Section 5.2.6.5 is not required.

3.2.4 <u>FSS Approach for Basement Structures</u>

The licensee provided additional information in the LTP Rev. 2 regarding FSS survey methodologies for basement structures and embedded piping to allow for increased flexibility. This includes discussion on the option of using traditional methods (hand-scanning and volumetric sampling for compliance) for surveys in basements, changes to how investigations will be conducted, and changes to remediation and resurvey levels. Further discussion and evaluation of these changes can be found in the subsections below.

3.2.4.1 FSS Preparation and Investigation Process

FCS LTP Rev. 2, Section 5.4.1.12, "Post-Demolition Surveys," now states that after demolition of Class 1 structures with basements when all areas are cleaned, a RASS will be performed using hand-held NaI or beta-gamma instrumentation to evaluate the need for additional remediation or to verify that FSS can proceed. The licensee states that areas greater than BcDCGL will be assessed against the $DCGL_{EMC}$ and areas that exceed the $DCGL_{EMC}$ will be remediated. Additionally, areas exceeding the O_PDCGL will be bounded and investigated using either ISOCS with reduced and/or overlapping FOVs to identify any elevated areas, or hand-held gamma scanning. Similarly, the LTP Rev. 2, Section 5.5.5.2, "Remediation, Reclassification, and Resurvey," states that for "Class 1 open land, basement and embedded pipe survey units, areas of elevated activity above the $DCGL_{EMC}$ will be remediated." If an area is remediated, then a RASS will be performed to ensure that the remediation is sufficient.

FC-24-004 contains a similar, but inconsistent, discussion on Investigation and Action Levels to be used during FSS. According to the TSD, investigation levels will be based on the Base Case DCGL (rather than the $DCGL_{EMC}$) for scanning with a NaI detector or ISOCS detector and onsite analysis. For a Class 1 survey unit, the Investigation Level is set at the Base Case DCGL. According to FC-24-004, any areas of elevated residual radioactivity above the Base Case DCGL (rather than the $DCGL_{EMC}$) will be remediated. If an area is remediated, then a RASS will be performed to ensure that the remediation is sufficient.

The licensee clarified during the NRC audit meeting (ML25154A358) that the language in FC-24-004 is incorrect. Investigation levels for Class 1 areas will be based on the $DCGL_{EMC}$, and remediation of Class 1 open land, basement, and embedded pipe survey units will occur for areas of elevated activity above $DCGL_{EMC}$. FCS LTP Rev. 2, Table 5-33, "Investigation Levels," shows scan and direct investigation levels that correspond to example FSS investigation levels found in MARSSIM Table 5.8, "Example Final Status Survey Investigation Levels."

3.2.4.2 FSS Measurement Methods for Basement Structures

FCS LTP Rev. 2, Section 5.4.1.1.2, "Gamma Scanning," now states that gamma scanning will be performed on basement surfaces using a Ludlum Model 44-10 NaI detector. The LTP Rev. 2, Section 5.3.5, "Sample Size Determination for FSS of Basement Walls/Floor When Using ISOCS," also states that either ISOCS or traditional methods (hand-held detector scanning and volumetric sampling) can be used to perform FSS on basement surfaces. This differs from the previous LTP Rev. 1, which stated that only ISOCS would be used to conduct FSS in basement structures.

3.2.4.3 Data Assessment for Basement Structures

The licensee states in FC-24-004 that compliance for the FCS basement structures will be demonstrated through the collection and analysis of systematic concrete core samples generated in a triangular grid from a random starting point. This is discussed further in the LTP Rev. 2, Section 5.5.4, "Data Assessment for FSS of Basement Walls/Floors Performed with ISOCS." Most of this section retains the language from the previous LTP Rev. 1 submittal but minor changes are included to (i) reflect that traditional hand-scanning/sampling will now be used in addition to ISOCS in basement areas, (ii) Area Factors along with elevated measurement concentrations (EMC) will now be implemented, and (iii) all Class 1 basement areas above $DCGL_{EMC}$ will be remediated rather than using the BcDCGL. The general approach outlined in the LTP Rev. 2, Section 5.5.4 is to collect a sufficient number of ISOCS measurements/concrete samples, record measured activity including background (background will not be subtracted), and then conduct the Sign test using the sum of fractions (SOF) for each measurement as the sum value for the test. For areas that exceed the BcDCGL, EMC applies.

3.2.4.4 Change to FSS Plan Below Structures

In this section the NRC staff evaluates the changes to the FSS plans for sampling below structures associated with the contamination in the three grouted areas covered by the changes introduced in FCS LTP Rev. 2. Specifically, the NRC staff consider the possibility that residual radioactivity may have spread to the natural environment below the structures prior to being identified during continuing characterization. For this evaluation, the staff uses descriptions of contamination and remediation at the three grouted areas provided by the licensee in Section 6 “Compliance with Radiological Criteria for License Termination,” of the LTP Rev.2; FC-25-001, Revision 0; FC-25-002, Revision 0; and FC-24-004, Revision 1. For this evaluation, the NRC staff treat the trenches at the two elevations in the Auxiliary Building separately.

For the west wall, tendon stressing gallery of the Containment Building, and trenches at the 971-ft (296-m) elevation in the Auxiliary Building, regardless of whether the contamination event existed during plant operation or occurred during decommissioning, the NRC staff conclude that contamination likely did not migrate into the natural environment below or adjacent to the building. This conclusion is based on the groundwater potential (i.e., hydraulic head) being higher outside the structure, because the locations are well below any recorded groundwater level, thus generally inducing flow into rather than out of the structure. Therefore, the NRC staff finds the FSS plan in Section 5.4.1.3.3, “Sampling of Subsurface Soils below Structure Basement Foundations,” of the LTP Rev. 2 acceptable for the west wall, tendon stressing gallery of the CB, and trenches at the 971-ft (296-m) elevation.

For the Auxiliary Building trenches at the 989-ft (301.5-m) elevation, the NRC staff notes that there was uncertainty in the timing of the release from the pipes into the concrete (i.e., during operation or decommissioning). In addition, the floor slab was thin due to pipe removal and remediation scabbling, cracked, and in the zone of a fluctuating water table. Therefore, the staff determined that the potential existed for releases to have occurred to the natural environment below slabs at an unknown point in time and for an unknown length of time. The licensee did not update the FSS plans in Section 5.4.1.3.3 from the LTP Rev. 1 to account for the three grouted areas. While the licensee’s description generically included sediment sampling below structures, the licensee included caveats for sampling plans (e.g., high groundwater levels or difficult to reach with boreholes) that left open the possibility that no sampling may be an outcome.

In Enclosure 2 of the NRC audit summary (ML25154A358), OPPD provided draft text in response to Topic 7, “Description of the Release from Pipes and Continuing Characterization.” OPPD stated in the response to Topic 7 that samples below the slabs of the Auxiliary Building and Containment Buildings would be obtained through utilization of one or more of the following technologies: (i) diagonal boring using drilling rigs offset from the structure to obtain sufficient angle to obtain samples from beneath the areas of interest, (ii) drilling or boring directly through the slab and backfill after decommissioning activities have been completed, and (iii) utilizing hydraulic inverters that would be installed prior to backfilling, which would facilitate collection of water samples below the structures. However, the NRC staff notes that the draft text for Topic 7 was not incorporated into the July 2025 supplement to the LTP Rev.2. Therefore, the staff expects to monitor this area as part of routine oversight at the FCS site and to address incorporation of this language into subsequent revisions of the FCS LTP, as necessary.

Considering OPPD's FSS plan for sampling below the slabs in areas tied to the grouted structures, particularly the additional measures planned for the Auxiliary Building trenches at the 989-ft (301.5-m) elevation, the NRC staff concludes that the FSS sub-slab sampling plan is acceptable because it will provide reasonable assurance that residual radioactivity below the structures has been adequately characterized and dose will not be underestimated.

3.2.4.5 NRC Evaluation of FSS Approach for Basement Structures

With regard to revisions to FSS preparation and the investigation process for Class I basement surfaces: because the revised investigation levels are based on $DCGL_{EMC}$ for scanning and either $DCGL_{EMC}$ or OpDCGL for direct measurements, and the licensee has committed to remediation of any areas of elevated residual radioactivity above $DCGL_{EMC}$, investigation levels will be triggered as needed for remediation. Furthermore, the investigation levels listed in the LTP Rev. 2, Table 5-33 correspond to example guidance in MARSSIM Table 5.8. Whether revised investigations consist solely of ISOCS measurements or hand-scanning plus additional ISOCS investigations, the NRC staff has adequate assurance that investigations will be able to locate areas of elevated radioactivity. As noted in Section 5.6.5.2, "NRC Evaluation of Final Status Survey Approach for Basement Structures," of the SER for the previous LTP Rev. 1 submittal (ML24019A168), preliminary demolition and RASS surveys could be used to locate areas of elevated activity in instances where ISOCS is used to conduct investigations, and in instances where hand scanning is utilized, the NRC staff maintain adequate assurance that these conventional scan surveys will be conducted to confirm and/or identify the location of discrete radioactive particles.

With regard to the FSS methods for Class I basement structures, the LTP Rev. 2 maintains language regarding the use of ISOCS but also includes typical hand scanning and sampling for certain parts of basement structures. Because the NRC previously reviewed and approved the use of ISOCS in basement structures in the SER for the previous LTP Rev. 1 submittal, and new language regarding typical hand scanning/sampling aligns with guidance in MARRSIM, the NRC staff maintain reasonable assurance that adequate remediation and characterization efforts will be made to assure basements will meet the unrestricted release criteria.

Based on its evaluation of the entire survey process for basement structures, the NRC staff finds the revised language describing the overall survey methodology consisting of either ISOCS or hand scanning and systematic sampling, 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 $DCGL_{EMC}$, consistent with the MARSSIM guidance regarding the radiological sampling approach and detection of elevated areas. The NRC staff has already evaluated the adequacy of ISOCS for performing FSS in basement areas in the SER for the previous LTP Rev. 1. Similar to the previous conclusion, the NRC staff finds that for basement areas where FSS is performed with hand scanning and systematic sampling as described in the LTP Rev. 2, the Ludlum 44-10 detection sensitivities being less than the easy-to-detect DCGLs, combined with either the surrogate ROC process or direct analysis of HTD ROC concentrations, should be adequate to detect residual radioactivity. The licensee's process for developing a SOF result from basement surfaces based on sample measurements for input into the Sign test is adequate to demonstrate compliance with the DCGLs and is therefore acceptable to demonstrate compliance with 10 CFR 50.82(a)(9)(ii)(D).

3.2.5 FSS Instrument Selection for Basement Structures

As discussed in Section 3.2.4 of the SER, a combination of gamma scanning using a Ludlum Model 44-10 or 44-20 NaI detector and ISOCS will be used to perform FSS on basement surfaces. As noted in the LTP Rev. 2, Section 5.4.2.1, "Instrument Selection," the instruments chosen to conduct scan measurements in Class 1 areas will be required to be capable of detecting radioactive material at the BcDCGL. However, the licensee does caveat that measurement results with associated MDCs that exceed these values may be accepted as valid data after evaluation by FSS supervision. The evaluation will consider the actual MDC, the reported value for the measurement result, the reported uncertainty, and the fraction of the OpDCGL identified in the sample. FCS LTP Rev. 2, Section 5.4.2.1 also maintains language that "for direct measurements and sample analyses, MDCs less than 10 percent of the OpDCGL are preferable while MDCs up to 50 percent of the OpDCGL are acceptable." TSD FC-24-003, "Ludlum 44-10 Detector Sensitivity for Determining Scan MDCs on Concrete Surfaces" (ML26027A169), presents the Ludlum Model 44-10 Scan MDC for concrete surfaces.

3.2.5.1 Gamma Scanning

The licensee states that the primary purpose of FC-24-003 is to evaluate the sensitivity of the Ludlum Model 44-10 2.0"x2.0" NaI detector when used to scan concrete surfaces of building structures as a function of background, scan speed, source dimensions, and radionuclide mixture. The methodology used follows guidance from MARSSIM Section 6.7.2.1, "Scanning for Beta and Gamma Emitters," and NUREG-1507, "Minimum Detectable Concentrations with Typical Radiation Survey for Instruments for Various Contaminants and Field Conditions" (ML20233A507), for determining the scan MDC for gamma emitting radionuclides. The TSD contains evaluations for determining detector FOV, source term depth, concrete density, instrument efficiency, and source efficiency, for input into NUREG-1507 Equation 6.11. The licensee specifies the detector height and scan rate for concrete surfaces in the LTP Rev. 2, Section 5.4.1.1.2. All parameters are also specified in FC-24-003.

3.2.5.2 In-Situ Gamma Spectroscopy

The licensee also submitted TSD FC-24-001, "Use of In-Situ Gamma Spectroscopy for Scan Surveys of Building Surfaces" (ML26027A168), which provides a description of the use and efficiency calibration of a Mirion In-Situ Object Counting System (ISOCS) for concrete surfaces. To perform ISOCS instrument efficiency calculations, the licensee used Genie-2000 software and took into account the areal distribution of the contamination (i.e. depth of contamination similar to FC-24-003), concrete density, and sensitivity to non-uniform areal distributions, which OPPD calculated by using offset geometry. Then, derived ISOCS investigation levels were established using the following equation:

Equation 1: ISOCS Investigation Levels

Nuclide Derived Investigation Level (pCi/m^2) = DCGL * CF (offset geometry correction factor)

Example investigation levels for Cs-137 and Co-60 are included in Table 5, "Correction Factor and Derived Investigation Level Calculation," of FC-24-001. The licensee states that the geometries must be reviewed by a qualified competent person to ensure they are correctly developed and accurately or conservatively represent the media being measured, and the qualifications of the competent person should be documented. The licensee also states that in

Class 1 survey units where 100 percent scan coverage is required, detector FOV between adjacent measurements will be overlapped to ensure that there are no gaps in coverage.

3.2.5.3 NRC Evaluation of FSS Instrument Selection for Basement Structures

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, "Final Status Survey Design," of NUREG-1757, Volume 2, Revision 2, and Section 2.5 of the SRP. The NRC staff verified *a priori* scan MDC for the Ludlum Model 44-10 in the LTP Rev. 2, Table 5-29, "Recommended Survey Coverage for Open Land Areas and Structures," and FC-24-003 for a range of scenarios using the inputs and equations provided in FC-24-003 and the LTP Rev. 2, Section 5.4.1.1.2. The NRC staff notes that the Ludlum Model 44-10 scan MDC taken from FC-24-003 and listed in the LTP Rev.2, Table 5-32, "Typical FSS Instrument Detection Sensitivities," are for a different scenario (a detector height of 2 inches with a background count rate of 4500 counts per minute(cpm)) than the typical scenario indicated by the licensee (a detector height of 6 in with a background count rate of 18000 cpm) during the audit (ML25154A358). However, the NRC staff reproduced the typical scan MDC given by the licensee during the audit clarification discussions for concrete surfaces, which resulted in a similar typical MDC value. The *a posteriori* scan MDC for these areas will be evaluated during review of the FSS release records. Regarding ISOCS, geometries used to calculate investigation levels will be reviewed by a competent person with documented qualifications in accordance with the LTP Rev. 2. Overall, the NRC staff finds that the proposed radiation detection and measurement instrumentation for performing FSS in the LTP Rev. 2 are consistent with MARSSIM and NUREG-1507. Therefore, the NRC staff finds the licensee's proposed FSS instrumentation in the LTP Rev. 2 acceptable to demonstrate compliance, in part, with 10 CFR 50.82(a)(9)(ii)(D).

3.3 Dose Modeling

The LAR to approve Revision 2 of the FCS LTP requested a revision to the scenarios and corresponding DCGLs associated with the Auxiliary Building basement to add additional flexibility to the methodologies needed to satisfy the applicable regulatory requirements. The supplement to the LAR requested consideration of additional DCGLs associated with portions of the Containment Building. These proposed changes require the reconsideration of the site conceptual model and the exposure scenarios used for evaluating whether resulting doses demonstrate compliance with the dose criterion for unrestricted release in accordance with 10 CFR 20.1402. Specific changes include:

1. Revise the Residential Farmer Basement Fill Model (BFM) excavation scenario by changing the assumption that concrete acts as a soil.

2. Modify the BFM excavation scenario for the Resident Farmer to include basement walls less than or equal to 3 meters below ground surface (bgs).

3. Inclusion of the Industrial Use BFM that addressed walls and floor greater than or equal to 3 meters bgs.

4. Change DCGL selection to year of highest total relative dose for Industrial Use BFM excavation scenario only.

5. Use of grout or concrete fill to account for reduction of radionuclide release to groundwater.

The LTP Rev. 2 also proposes the addition of site-specific DCGLs for two areas of the Containment Building: the tendon stressing gallery and a portion of the west wall at the 973-ft (296.6-m) elevation.

OPPD proposed in the LAR, as supplemented, amending Chapter 6 of the FCS LTP to incorporate these five changes, which they indicated will reduce conservatism: The following sections provide the NRC staff's evaluation of these requested changes.

3.3.1 Updating the Previously Approved Resident Farmer Exposure Scenarios

OPPD is proposing to modify the exposure scenarios for sections of the Auxiliary Building and Containment Building from the scenarios approved in the SER for the LTP Rev. 1. Although conservatism in dose modeling is acceptable, NRC guidance in NUREG-1757, Volume 2, Revision 2, allows for the use of site-specific parameters whenever possible, provided the values can be justified. Taking this guidance into account, OPPD proposes to modify the previously approved Resident Farmer BFM excavation scenario described in Section 6.11.2, "BFM Wall/Floor Conceptual Model, Source Term Abstraction and Release," of the LTP Rev. 1, which assumes that concrete brought to the surface by excavation activities has the same capacity to grow plants as soil.

Section 6.13.1, "Auxiliary Building BFM Excavation Scenario for Resident Farmer," of the LTP Rev. 2 proposes to differentiate concrete from soil and consider two Resident Farmer BFM excavation scenarios.

- The first scenario assumes that concrete is excavated and used as a 1-meter (1-m) layer of fill on the site with no soil cover. This approach assumes that plants and fodder cannot establish roots directly into the concrete and thus, the plant, meat, and milk pathways associated with the scenario are not considered.

- The second scenario, like the first, assumes that concrete is excavated and used as a 1-m layer of fill on the site. However, this scenario also considers an additional 1-m of soil as a cover on top of the concrete fill. Adding the layer of soil allows for plants and fodder to be grown on the site, resulting in the plant, meat, and milk exposure pathways being considered.

OPPD would then select the smaller of the two DCGL values as the Resident Farmer excavation scenario DCGL.

OPPD states that this approach would reduce conservatism associated with the assumption that concrete acts like soil and would apply more realistic Resident Farmer exposure scenarios when evaluating compliance with regulatory requirements. The first updated Resident Farmer excavation scenario eliminates the consideration of the growth of plants and fodder and therefore negates the inclusion of the plant ingestion, meat ingestion, and milk ingestion exposure pathways. The addition of a 1-m layer of soil on top of the concrete associated with the second updated Resident Farmer excavation scenario allows for growth of plants and fodder and the inclusion of the plant ingestion, meat ingestion, and milk ingestion exposure pathways typically included with a resident farmer scenario.

OPPD also proposes to limit the use of the Resident Farmer scenario to activities performed less than 3 meters bgs. This change is discussed further in Section 3.3.2 of the SER. NUREG-1757, Volume 2, Revision 2, notes that residential basements are assumed to be dug to a depth of 3 meters bgs. Plant depth details provided in ANL/EVS/TM-14/4, "Data Collection Handbook to Support Modeling Impacts of Radioactive Material in Soil and Building Structures," indicate that plants from which nutrients are obtained generally have root depths that are less than 1-m bgs. Additionally, the RESRAD-ONSITE default parameter for root depth is 0.9 meter and allows a site-specific value between 0.3 and 4.0 meters. Considering these details, the NRC staff found the basis for using the updated resident farmer exposure scenarios to be acceptable. Given these considerations, the NRC staff also considered the proposal to limit the use of Resident Farmer DCGLs when evaluating the top 3 meters bgs to be acceptable.

3.3.2 Incorporating an Industrial Use Scenario

OPPD notes in Section 6.13.2, "Auxiliary Building Basement Resident Farmer Wall Excavation," of the LTP Rev. 2 that excavation activities greater than 3 meters bgs are inconsistent with the Resident Farmer scenario. OPPD therefore proposes to limit the Resident Farmer scenario to excavation activities associated with the removal of walls in the top 3 meters bgs and to use an Industrial Use scenario when evaluating material excavated from greater than 3 meters bgs. OPPD considers the Industrial Use exposure scenario to be a more realistic, yet still reasonably conservative, compliance scenario for evaluating the Auxiliary Building basement walls and floors and trenches located more than 3 meters bgs. The Industrial Use scenario provided in the LTP Rev. 2 considers pathways associated with excavation activities as well as in situ and drilling spoils exposure scenarios.

The NRC staff found the incorporation of an Industrial Use scenario for depths greater than 3 meters bgs to be acceptable when evaluating contamination associated with the Auxiliary Building basement because use of the Resident Farmer scenario below that depth is less realistic and use of the Industrial Use scenario remains conservative. As previously indicated, NUREG-1757, Volume 2, Revision 2, notes that residential basements are assumed to be dug to a depth of 3 meters bgs, so it is unlikely that a resident farmer would be exposed to material located at depths greater than 3 meters bgs. The NRC staff noted that although an Industrial Use scenario was used to evaluate doses associated with material being excavated from greater than 3 meters bgs, the final DCGLs used for evaluating compliance are based on the excavated material being placed on the surface. Therefore, the DCGLs calculated using the Resident Farmer exposure scenario are still considered when evaluating compliance.

3.3.3 Considering a Modified Resident Farmer Scenario – NRC Independent Analysis

FCS LTP Rev. 1 and corresponding SER considered the current land use around the site to be agricultural, though industrial activities to the north, along the river, suggest that both land uses could occur in the future. The licensee implicitly acknowledges both uses in their development of DCGLs. However, when considering the excavation of material from greater than 3 meters bgs, the licensee only assumes that the site will be used for industrial use. No considerations were provided for the possibility that farming occurs adjacent to where the deeper excavated concrete has been brought to the surface.

The NRC staff considered a modified Resident Farmer scenario to address the possibility of a resident farmer being exposed to Auxiliary Building basement material excavated from greater than 3 meters bgs. To calculate applicable DCGLs, the NRC staff used the exposure pathways considered for the two newly proposed Resident Farmer scenarios, as well as the soil to concrete DCGL conversion factors based on the surface area to volume ratio for the Auxiliary Building walls and floors associated with the newly proposed Industrial Use exposure scenario. The resulting concrete excavation DCGLs calculated using this exposure scenario were larger than the Resident Farmer DCGLs and, thus, are bound by the values proposed for the Resident Farmer concrete excavation DCGLs.

3.3.4 Considering Exposure to the Containment Building West Wall and Stressing Gallery

Since approval of the LTP Rev. 1, two areas of the Containment Building were determined to be suitable for grouting: a portion of the Containment Building West Wall at the 973-ft (296.6-m) elevation and the Tendon Stressing Gallery. A final status survey, and thus site-specific concrete excavation DCGLs would be required prior to the grouting of each of these areas.

Section 6.14, “Containment Building 973-ft Elevation West Wall Activated Concrete DCGLs,” of the LTP Rev. 2 discusses OPPD’s approach for selecting DCGLs for the 973-ft (296.6-m) elevation Containment Building West Wall that was subject to neutron activation. A photograph of the area is provided in Figure 6-19, “973-foot West Wall Containment Area of Activated Concrete,” of the LTP Rev. 2. OPPD noted that characteristics of this area are essentially the same as those used for the grouted east trenches of the Auxiliary Building discussed in Section 6.13, “BFM Wall/Floor DCGLs for the Auxiliary Building Basement,” of the LTP Rev. 2.

The Stressing Gallery, an unlined concrete structure, is located below the liner under the Containment Building. Specific details are provided in Section 6.15, “BFM Wall/Floor DCGLs and Excavation Scenario DCGLs for the Stressing Gallery,” of the LTP Rev. 2. Because the previously developed Containment Building DCGL values consider the presence of a liner, it is reasonable to assume that the concrete associated with the Stressing Gallery is clean, warranting separate excavation DCGLs that only consider concrete.

3.3.5 Updating Scenario-Specific DCGLs

Modifications to the Resident Farmer exposure scenario and the addition of an Industrial Use exposure scenario required OPPD to update the previously approved radionuclide-specific DCGLs and develop new scenario-specific DCGLs for use when evaluating compliance for the Auxiliary Building basement and east trenches. DCGLs for specific areas of the Containment Building were also provided in the LTP Rev. 2. This section summarizes the changes to previously approved DCGLs, as well as the DCGLs included in OPPD’s supplemental analyses.

3.3.5.1 Updated Resident Farmer DCGLs

No changes were made to the Resident Farmer drilling spoils and in situ DCGLs previously approved in the SER for Revision 1 of the FCS LTP.

Section 6.13.1 of the of the LTP Rev. 2 discusses revisions to the previously approved Resident Farmer excavation DCGLs using the updated Resident Farmer scenarios intended to provide “…two different, but more reasonable, excavated concrete geometries.” OPPD calculated new radionuclide-specific soil DCGLs for the two Resident Farmer excavation scenarios. A “unit concentration in excavated concrete” value based on the density of concrete (2.2 grams per cubic centimeter), as well as the surface area to volume ratio for the Auxiliary Building, calculated using Equation 6-8 of the LTP Rev. 2, was used to convert soil DCGLs to excavated concrete DCGLs using Equation 6-9 from the LTP Rev. 2. The NRC staff evaluated OPPD’s RESRAD-ONSITE, Version 7.2, analyses and the conversion to Resident Farmer concrete excavation DCGLs using Equations 6-8 and 6-9 and found the approach acceptable for determining radionuclide-specific concrete excavation DCGL values for both scenarios.

In response to the audit review RAIs (ML25211A290), OPPD noted that the 0.95 adjustment factor used to determine the original Resident Farmer DCGL values for use when considering doses associated with walls in the top 3 meters bgs was unnecessary and caused the sum of fractions calculation to be greater than 1.0. Therefore, OPPD removed the adjustment factor and recalculated the Resident Farmer DCGL values in the LTP Rev. 2. The NRC staff found that this change, along with the use of more site-specific Resident Farmer scenarios, to be acceptable even though it resulted in an increase in the DCGL values from those approved in the LTP Rev. 1 because the parameters used were more realistic to the scenario evaluated.

Table 6-19, “Resident Farmer Auxiliary Basement BFM Excavation Scenario DCGLs,” in the LTP Rev. 2 compares the Resident Farmer excavation DCGLs for both scenarios. The NRC staff agree with OPPD’s proposal to use the smaller of the two adjusted radionuclide-specific DCGL values as the assigned Resident Farmer BFM concrete excavation scenario DCGL value. The staff noted that by selecting the smallest of the two Resident Farmer concrete excavation DCGL values, 10 of the 24 radionuclides selected are associated with a Resident Farmer scenario that does not include exposure pathways typically considered when evaluating a Resident Farmer scenario. Although it is reasonable to assume that these exposure pathways would not exist on a site with a concrete surface, the NRC staff performed an independent analysis using a “Modified Resident Farmer” scenario that includes a concrete surface as well as the plant, meat, and milk ingestion pathways to evaluate the impacts from removing these pathways. The “Modified Resident Farmer” DCGL values were larger (i.e., less conservative) than the DCGLs calculated for at least one of the proposed Resident Farmer scenarios. Based on these findings, the NRC staff concluded OPPD’s approach for selecting Resident Farmer DCGLs using the two proposed scenarios was conservative, and therefore acceptable.

While evaluating the RESRAD analyses, the NRC staff also noted that the licensee eliminated the plant, meat, and milk exposure pathways associated with the scenario that included a 1-m layer of concrete on the surface by setting the contamination fractions for each of these pathways to zero instead of turning off the pathways using the “Set Pathways” function in the RESRAD menu. The NRC staff performed an independent analysis to confirm that this approach produced the same results as turning off the specific pathways in the “Set Pathways” menu and that no additional parameter values were impacted. Results from this analysis confirmed that both approaches provided the same results.

3.3.5.2 Auxiliary Building-Related Industrial Use DCGL Values

Industrial Use scenario DCGLs for the Auxiliary Building were not previously considered in the LTP Rev. 1. The LTP Rev. 2 proposes the use of separate Industrial Use DCGLs for the Containment Building 973-ft (296.6-m) West Wall and the Stressing Gallery.

The NRC staff reviewed OPPD’s approach for deriving radionuclide-specific concrete in situ, drilling spoils, and excavation DCGLs for walls and floors, as well as the east trenches associated with the Auxiliary Building, from soil DCGLs calculated using RESRAD, Version 7.2, and found it to be acceptable. This included reviewing (i) the changes to RESRAD parameter values for the Industrial Use scenario provided in Table 6-20, “Parameter Changes for Industrial Use Soil DCGLs,” of the LTP Rev. 2; (ii) the approach used for deriving Industrial Use in situ and drilling spoils DCGLs discussed in Section 6.11.6, “BFM *in situ* Scenario Initial Suite DCGLs,” and Section 6.11.7, “BFM Drilling Spoils Scenario Mathematical Model.” of the LTP

Rev. 2, respectively; and (iii) the use of the “unit concentration in excavated concrete” value derived using Equation 6-8 of the LTP Rev. 2 for converting Industrial Use soil DCGLs to Industrial Use concrete excavation DCGLs. While evaluating the approach for deriving these DCGLs, the NRC staff noted that OPPD used a concrete density value of 1.49 grams per cubic centimeter when calculating Industrial Use in situ and drilling spoils DCGLs and a concrete density value of 2.2 grams per cubic centimeter, a value commonly used to represent concrete density, when calculating Industrial Use concrete excavation DCGL values. No basis was provided for the use of different concrete density values. Independent analyses performed using both densities demonstrated that smaller concrete density values resulted in more restrictive DCGL values. Therefore, the NRC staff find it acceptable for OPPD to use DCGL values calculated using smaller concrete densities for in situ and drilling spoil DCGLs.

OPPD’s approach for selecting Auxiliary Building Industrial Use compliance DCGLs included the use of a radionuclide-specific concrete excavation DCGL for both the Auxiliary Building walls and floors as well as the east trenches that correspond to the year with the highest relative dose. This approach differs from the previously approved approach of selecting the radionuclide-specific peak dose DCGL calculated using RESRAD as discussed in Section 2.7 of NUREG-1757, Volume 2, Revision 2. The proposed approach of dividing the mixture fraction value, which corresponds to a hypothetical concentration for each radionuclide at year zero, by the concrete excavation DCGL results in a radionuclide-specific relative dose value based on the fraction of the specific radionuclide that currently exists in the excavated concrete. The relative dose values are then summed for each year considered in the RESRAD analysis to determine which specific year has the largest total relative dose for all radionuclides included in the mixture. Using this approach, the chosen year for both the excavation scenario wall/floor DCGLs and trench DCGLs was determined to be year zero. As a result, the Industrial Use concrete excavation DCGLs calculated using the RESRAD analysis and Equation 6-9 of the LTP Rev. 2 for year zero were selected as the Industrial Use concrete excavation DCGLs, and ultimately as the compliance DCGLs used when evaluating most of the radionuclides associated with the concrete excavated from the Auxiliary Building basement walls and floors and east trenches. The NRC staff agree that this approach is more realistic as it enables the licensee to use DCGLs associated with a single year instead of using the conservative, bounding approach that involves using radionuclide-specific DCGLs associated with several peak dose years.

RESRAD analyses determined that 6 of the 24 radionuclides being evaluated had peak doses occurring at years other than year zero (Table 1). Five of these radionuclides had DCGL values at year zero that were comparable to their peak year DCGL value. Current concentrations of Neptunium-237 (Np-237), the sixth radionuclide, have been measured at concentrations well below the year zero DCGL value; increases in Np-237 concentrations are associated with ingrowth over time will not be of concern at its peak dose year (341).

Table 1. Comparison of relative dose year and peak dose year for radionuclides with peak dose years not at year zero

Radionuclide	Relative Dose Year	DCGL	Peak Dose Year	DCGL
		pCi/m2		*pCi/m2*
H-3	0	3.442E+11	2.069	1.376E+11
Ni-59	0	9.391E+13	1000	1.224E+13
Np-237	0	4.541E+08	341.4	4.890E+06
Pu-239	0	5.325E+09	1000	5.198E+09
Pu-241	0	3.645E+11	39.75	1.256E+11

Tc-99	0	4.356E+10	39.75	1.277E+09

3.3.5.3 Containment Building-Related DCGLs

In the LTP Rev. 2, OPPD proposed site-specific Industrial Use DCGL values for two areas of the Containment Building: a portion of the Containment Building concrete wall at the 973-ft (296.6-m) elevation, identified as the West Wall, and the Stressing Gallery.

3.3.5.3.1 DCGLs for 973-Foot Elevation West Wall

The NRC staff reviewed and found it acceptable for OPPD to use the Auxiliary Building DCGL values provided in Table 6-23, "Auxiliary Building Basement BFM Scenario DCGLs for Resident Farmer and Industrial Use," of the LTP Rev. 2, for the Containment Building concrete 973-ft (296.6-m) elevation West Wall (see Table 6-26, "Containment Concrete 973-Foot Elevation West Wall BFM Scenario DCGLs for Resident Farmer and Industrial Use") for the Resident Farmer and Industrial Use in situ and drilling spoils scenarios. The NRC staff also agree that there is no need to consider DCGLs for the Resident Farmer excavation scenario since the 973-ft (296.6-m) elevation West Wall is located more than 3 meters bgs and excavation at that depth is not considered to be a reasonable activity for resident farmers.

Using the details provided by OPPD, the NRC staff calculated similar Industrial Use concrete excavation DCGL values using the Containment Building "effective wall thickness" provided in OPPD's FC-20-006, Revision 0, Equation 6-8 of the LTP Rev. 2 to calculate the "unit concentration in excavated concrete," and Equation 6-9 of the LTP Rev. 2 to convert the Industrial Use soil DCGLs calculated using RESRAD, Version 7.2, to Industrial Use concrete excavation DCGLs. Table 6-27, "DCGLs Applicable to Grouted Containment 973-Foot Elevation West Wall Concrete," of the LTP Rev. 2 provides the applicable DCGLs for each scenario calculated using the DCGL values in Table 6-26 of the LTP Rev. 2 and Equations 6-23, 6-24, and 6-25 of the LTP Rev. 2. The final DCGLs for the grouted Containment Building 973-ft (296.6-m) elevation West Wall, which are the smallest radionuclide value of the three provided in Table 6-27 for each radionuclide, are provided in Table 6-28, "Final DCGLs For Grouted Containment Building 973-Foot Elevation West Wall Concrete," of the LTP Rev. 2.

3.3.5.3.2 DCGLs for Tendon Stressing Gallery

Section 6.15 of the LTP Rev. 2 discusses the development of DCGLs for the Stressing Gallery, an unlined concrete structure below but connected to the Containment Building. Although DCGL values were developed for the Containment Building (Table 5-10, "OpDCGLs for Basement Floor/Walls ($OpDCGL_{wf}$) Adjusted for IC Dose," of the LTP Rev. 2), the NRC staff agree that separate Stressing Gallery DCGLs are appropriate given that previously calculated Containment Building BFM excavation DCGLs are based on the presence of the liner, all concrete above the liner will be excavated, and, aside from the 973-ft (296.6-m) elevation west wall, all concrete remaining in the Stressing Gallery below the liner is assumed to be clean.

OPPD calculated the final Stressing Gallery DCGLs using Equation 6-13 of the LTP Rev. 2, which considers in situ DCGLs, drilling spoils DCGLs, and excavation DCGLs. The Containment Building in situ DCGLs were used after being adjusted for diffusion, the Containment Building drilling spoils DCGLs were applied unchanged, and the excavation DCGLs were calculated as described above and listed in Table 6-29, "Stressing Gallery Soil DCGLs and Corresponding Excavation DCGLs," of the LTP Rev. 2. The Stressing Gallery wall/floor DCGLs provided by OPPD in Table 6-30, "Wall/Floor DCGLs for the Stressing Gallery," of the LTP Rev. 2 are more conservative (i.e., smaller values) than the values calculated by the NRC staff and therefore acceptable for use because they represent more conservative values.

3.3.5.4 Summary of Compliance DCGL Values

This section lists the tables in the LTP Rev. 2 containing the updated Auxiliary Building DCGLs, as well as the DCGL values for the 973-ft (296.6-m) West Wall and the Stressing Gallery.

- Table 6-18, "BFM Wall/Floor Initial Suite DCGLs (No IC Dose Correction)," provides the wall/floor DCGL values for the Containment Building (not including the West Wall and Stressing Gallery).
- Table 6-25, "Auxiliary Building Basement BFM Wall/Floor and Trench DCGLs," provides the Auxiliary Building basement wall/floor and East Trench DCGLs.
- Table 6-28 provides the final DCGLs for the grouted Containment Building 973-ft (296.6-m) elevation West Wall concrete.
- Table 6-30 provides the final DCGL values for the wall/floor of the Stressing Gallery.

3.4 Source Release for Grouted Areas

In Sections 6.13, 6.14, and 6.15 of the LTP Rev. 2, FC-24-002, "Radionuclide Release Through Grouted Trenches," Revision 0 (ML26027A170), and as discussed in OPPD's Response to Audit Question 6 (ML26033A033), the licensee revised the BFM in situ scenario to account for the reduction of radionuclide release from the three grouted areas by applying a diffusion-based model. Previously, in the LTP Rev. 1, the licensee used an instantaneous release model from the contaminated concrete and assumed the radionuclides were fully mixed in an idealized rectilinear 1-m thick backfill zone conceptually adjacent to cement walls and floors. The NRC staff evaluated the development of the grouted BFM In Situ and Drill Spoils DCGLs in Section 3.3 of this SER. In this section of the SER, the NRC staff will evaluate the calculation of the diffusion-based fractional cumulative release for each radionuclide in the entire suite of radionuclides provided Table 6-22, "Cumulative Fractional Release from Uniformly Contaminated Grout of 45 cm with 30 cm Cover," of the LTP Rev. 2.

The licensee indicated in Section 6.13.4, "BFM in situ DCGLs for Auxiliary Building Trenches after Filling with Grout or Concrete," of the LTP Rev. 2 that the emplaced grout, together with the relatively low permeability of existing concrete, would limit groundwater intrusion into or out of the Auxiliary Building basement, thus acting as an engineered barrier to reduce the migration rates of contamination. In addition, the licensee injected grout into visible cracks and injected grout below the slab at trench locations, which reduced ongoing groundwater influx and reduced potential future groundwater flux between the natural sediments below the slabs and the trenches. With the reduction in advective flow rates of water, the migration of radionuclides becomes dominated by the diffusion process.

The licensee described in Section 6.13 of the LTP Rev. 2 how the grout-related DCGLs incorporated the effect of diffusion. The licensee calculated cumulative diffusion release factors over a 1,000-year period for each radionuclide from the results of simulations using the DUST-MS model.[1,2] The licensee derived the factors in Table 5-2 of FC-24-002, Revision 0,

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1 Sullivan, T.M., "DUST - Disposal Unit Source Term: Data Input Guide." NUREG/CR-6041, BNL-NUREG-52375, 1993.

2 Sullivan, T.M., "DUSTMS_D - Disposal Unit Source Term Multiple Species - Distributed Failure Data Input Guide, Revision 1," BNL-75554-2006, Brookhaven National Laboratory, Upton, NY, 11973, January 2006.

based on simulations with a uniformly contaminated domain for the grout or concrete and a 30 centimeter (cm) (1-ft) thickness of clean grout cover, which is discussed further in the next paragraph on the model conceptualization. The licensee also multiplied the values from the DUST-MS results in Table 5-2 of FC-24-002, Revision 0, by a factor of three to produce the values of cumulative fractional release in Table 6-22 of the LTP Rev. 2; the factor of three accounted for the floor and two walls of the trenches all contributing to total contamination. Table 6-22 of the LTP Rev. 2 showed that only tritium, cesium, nickel, strontium, and technetium had fractional cumulative release values above 1×10^{-30} with tritium the highest at 0.147. As illustrated by Equations 6-16, 6-23 and 6-25 of the LTP Rev. 2, the cumulative diffusion release factors modify the RESRAD-derived DCGLs to estimate the grout-related DCGLs. The NRC staff notes that OPPD's calculation of RESRAD-derived DCGLs utilized sorption coefficients representative of sediments rather than that of grout or concrete, which leads to smaller DCGLs and is thus conservative.

The licensee described the DUST-MS conceptualization, model inputs, and simulation results in FC-24-002, Revision 0. The licensee calculated a cumulative fractional release over a 1,000-year period for each geometric configuration and initial and boundary conditions. The DUST-MS simulations implemented a one-dimensional model and included sensitivity that considered three different lengths over which diffusion would occur in the grout or concrete [i.e., 30, 45, and 90 cm (1, 1.5 and 3 ft)] and two versions of initial and boundary conditions. From FC-24-002, Revision 0, the licensee based the diffusion calculations on the grout thickness capping the grouted trenches in the Auxiliary Building, where that capping included the vertical thickness of grout covering the trenches and the horizontal distance of grout overlapping the edges of the trenches. The licensee provided geometric support in Sections 6.14 and 6.15 of the LTP Rev. 2 that the diffusion distance based on the trench grouting geometry conservatively covered the diffusion distances for the two grouted areas in the Containment Building. Therefore, the licensee applied the DUST-MS diffusion results into the grout-related DCGLs for each radionuclide for all three grouted areas. For sensitivity to the initial and boundary conditions, simulations included both (i) a clean grout or concrete domain with the radionuclide source as a boundary condition in the concrete surface, and (ii) uniformly mixed contamination as initial conditions across the domain in the grout or concrete. The licensee selected results from the conservative case (i.e., contamination uniformly distributed in grout within the trench and clean grout cover) for use in the DCCL calculations. The NRC staff note that this selection is conservative because the diffusion distance is shorter for the uniformly distributed contamination case.

The NRC staff notes that the selection of diffusion coefficients can significantly affect DUST-MS results. In DUST-MS, a diffusion-based process is simulated using effective diffusion coefficients. The licensee selected diffusion coefficients from ranges of laboratory measured values provided in the literature for a variety of grouts. The licensee chose values from the higher end of the literature-based range, which staff note is appropriate considering that the licensee did not provide the grout type and emplacement technique in the LTP Rev. 2 or supporting documents. Whereas diffusion coefficient values when measured and reported in the literature may be referred to as effective diffusion coefficients because they factor in effects due to water content, microcracks, and chemistry (e.g., mineral precipitation), the literature values may not include effects pertinent to the field scale where enhanced diffusion rates may occur; e.g., large scale fracturing that may lead to localized advective migration of radionuclides and degradation of grout or concrete over time. Noting that effective diffusion coefficients based on laboratory measurements can be one to two orders of magnitude larger than those measured in the laboratory on small samples, the NRC staff considered other literature.

For example, Figure A3-5 in the Center for Nuclear Waste Regulatory Analyses (CNWRA) (2006)[3] showed larger values of diffusion coefficients for radionuclides of carbon, cesium, cobalt, strontium, and technetium compared to the selections in Table 5.1 of FC-24-002, Revision 0. In addition, a report produced for the Zion Nuclear Power Station (BNL, 2016)[4] contained a table of selected diffusion coefficient values similar to the licensee's Table 5.1 in FC-24-002, Revision 0. The NRC staff noted the former included larger values of diffusion coefficients compared to those OPPD selected for the radionuclides of carbon, curium, neptunium, plutonium, and technetium. However, OPPD (see Response to Audit Question 6) noted that diffusion coefficients for those elements were set to that of cesium for the Zion backfill material, which OPPD stated is inappropriate for the intact grout and concrete at FCS, thereby justifying the different values in Table 5.1 of FC-24-002, Revision 0. The staff note that the potential underestimation of diffusion coefficients due to the scale effects described above is likely offset by the conservativeness of using cumulative diffusion over a 1,000-year period instead of a diffusion rate to reflect the effect of diffusion in radionuclide migration through grout.

Based on the discussion above, the NRC staff find that the estimates of the diffusion factor do not lead to underestimation of dose for the three grouted areas described in the LTP Rev. 2 because the (i) release of radionuclides through grout and concrete is dominantly a diffusion process, (ii) inputs are appropriate or conservative for all three grouted areas, (iii) the licensee selected diffusion coefficients at the higher end of the range from literature values, (iv) sorption coefficients used for the source release model in RESRAD simulations are smaller (and likely conservative) compared to sorption coefficients representative of grout and concrete, and (v) use of the cumulative effect of diffusion over a 1,000-year period is conservative and likely offsets the possibility of effective diffusion coefficients being larger than those from literature laboratory-based ranges of diffusion coefficients.

4 EPA CONSULTATION

The NRC originally engaged in a Level 1 consultation via letter with the U.S. Environmental Protection Agency (EPA) during the review of FCS LTP Rev. 1 (ML23082A220) because the licensee's proposed DCGLs for certain radionuclides at the site exceed the soil concentration values in Table 1 of the Memorandum of Understanding (ML022830208) between the NRC and the EPA. By letter dated September 28, 2023 (ML23276A004), the EPA responded to the NRC Level 1 consultation letter by providing its perspective and comments on the licensee's proposed DCGLs. The EPA also documented its understanding that the remediation activities at FCS associated with the NRC's decommissioning process are likely to significantly decrease the levels of those radionuclides present to residual levels below the DCGLs.

The LTP Rev. 2 did not propose any new soil DCGLs.Because the licensee is not changing their request for unrestricted release and groundwater conditions have not changed since the previous LTP Rev. 1 review, the NRC will not be re-engaging in a Level 1 consultation with the EPA under the MOU for the revision to the FCS license termination request.

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3 CNWRA, 2006, "Review of Literature and Assessment of Factors Relevant to Performance of Grouted Systems for Radioactive Waste Disposal," CNWRA 2009-001, Center for Nuclear Waste Regulatory Analyses. ML090980426.

4 BNL, 2016, "Brookhaven National Laboratory: Evaluation of Maximum Radionuclide Groundwater Concentrations for Basement Fill Model," TSD 14-009. ML16211A379.

5 STATE CONSULTATION

In accordance with the Commission's regulations, the Nebraska Department of Health and Human Services was notified of the proposed issuance of the amendment by email on December 2, 2025 (ML25342A008). The notification described the NRC's intent to issue a license amendment for the Fort Calhoun Station, Unit 1, to approve Revision 2 of the License Termination Plan and associated supplements dated October 2, 2024; January 16, 2025; July 28, 2025, July 31, 2025, and September 4, 2025. The state official had no comments.

6 ENVIRONMENTAL CONSIDERATION

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 pre-publication of 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 December 12, 2025 (90 FR 57788). 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.

7 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 Rev. 2 (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.