CHAPTER 12

RADIATION PROTECTION

TABLE OF CONTENTS

Section	Title	Page
12.1	ASSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS-LOW-AS-REASONABLY ACHIEVABLE (ALARA)	12.1-1
12.1.2.4.3	Equipment Layout	12.1-1
12.1.3	COMBINED LICENSE INFORMATION	12.1-1
12.2	RADIATION SOURCES	12.2-1
12.2.1.1.10	Miscellaneous Sources	12.2-1
12.2.1.3	Sources for the Core Melt Accident	12.2-3
12.2.1.3.1	Containment	12.2-3
12.2.1.3.2	Main Control Room HVAC Filters	12.2-3
12.2.3	COMBINED LICENSE INFORMATION	12.2-4
12.3	RADIATION PROTECTION DESIGN FEATURES	12.3-1
12.3.2.2.7	Control Room Shielding Design	12.3-1
12.3.4	AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION	12.3-1
12.3.5.1	Administrative Controls for Radiological Protection	12.3-5
12.3.5.2	Criteria and Methods for Radiological Protection	12.3-5
12.3.5.3	Groundwater Monitoring Program	12.3-5
12.3.5.4	Record of Operational Events of Interest for Decommissioning	12.3-5
12.4	DOSE ASSESSMENT	12.4-1
12.4.1.9	Dose to Construction Workers	12.4-1
12.4.1.9.1	Site Layout	12.4-1
12.4.1.9.2	Radiation Sources	12.4-1
12.4.1.9.3	Construction Worker Dose Estimates	12.4-2
12.4.1.9.4	Compliance with Dose Regulations	12.4-2
12.4.1.9.5	Collective Doses to LNP Unit 2 Workers	12.4-3
12.4.1.9.6	Operating Unit Radiological Surveys	12.4-3
12.5	HEALTH PHYSICS FACILITIES DESIGN	12.5-1
12.5.4	CONTROLLING ACCESS AND STAY TIME	12.5-1
12.5.5	COMBINED LICENSE INFORMATION	12.5-1

TABLE OF CONTENTS (Continued)

Section	Title	Page
APP. 12AA	RADIATION PROTECTION PROGRAM DESCRIPTION	12AA-1
12AA.5.4.14	Groundwater Monitoring Program	12AA-3
12AA.5.4.15	Record of Operational Events of Interest for Decommissioning	12AA-4

LIST OF TABLES

Number	Title
12.2-201	Core Melt Accident Source Strengths from MCR HVAC Filters as a Function of Time
12.2-202	Core Melt Accident Integrated Source Strengths from MCR HVAC Filters
12.4-201	Comparison of LNP Construction Worker Estimated Radiation Doses to 10 CFR 20.1301 Public Dose Criteria
12AA-201	Very High Radiation Areas (VHRA)

LIST OF FIGURES

Number	Title
12.3-201	Radiation Zones, Normal Operation/Shutdown, Nuclear Island, Elevation 100′-0″ & 107′-2″

CHAPTER 12

RADIATION PROTECTION

12.1 ASSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS-LOW-AS-REASONABLY ACHIEVABLE (ALARA)

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

STD COL 12.1-1

This section incorporates by reference NEI 07-08A, Generic FSAR Template Guidance for Ensuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA), Revision 0. See Table 1.6-201. ALARA practices are developed in a phased milestone approach as part of the procedures necessary to support the Radiation Protection Program. Table 13.4-201 describes the major milestones for ALARA procedures development and implementation.

Revise the last sentence of NEI 07-08A Subsection 12.1.2 to read:

STD COL 12.1-1

ALARA procedures are established, implemented, maintained and reviewed consistent with 10 CFR 20.1101 and the quality assurance criteria described in Part III of the Quality Assurance Program Description, which is discussed in Section 17.5.

Add the following information at the end of DCD Subsection 12.1.2.4:

12.1.2.4.3 Equipment Layout

STD SUP 12.1-1

A video record of the equipment layout in areas where radiation fields are expected to be high following operations may be used to assist in ALARA planning and to facilitate decommissioning.

12.1.3 COMBINED LICENSE INFORMATION

STD COL 12.1-1

This COL item is addressed in NEI 07-08A and Appendix 12AA.

12.2 RADIATION SOURCES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

12.2.1.1.10 Miscellaneous Sources

Add the following information at the end of DCD Subsection 12.2.1.1.10:

STD COL 12.2-1 Licensed sources containing byproduct, source, and special nuclear material that warrant shielding design consideration meet the applicable requirements of 10 CFR Parts 20, 30, 31, 32, 33, 34, 40, 50, and 70.

There are byproduct and source materials with known isotopes and activity manufactured for the purpose of measuring, checking, calibrating, or controlling processes quantitatively or qualitatively.

These sources include but are not limited to:

- Sources in field monitoring equipment.
- Sources in radiation monitors to maintain a threshold sensitivity.
- Sources used for radiographic operations.
- Depleted uranium slabs used to determine beta response and correction factors for portable monitoring instrumentation.
- Sources used to calibrate and response check field monitoring equipment (portable and fixed).
- Liquid standards and liquids or gases used to calibrate and verify calibration of laboratory counting and analyzing equipment.
- Radioactive waste generated by the use of radioactive sources.

Specific details of these sources are maintained in a database on-site following procurement. This database, at a minimum, contains the following information:

- Isotopic composition
- Location in the plant
- Source strength
- Source geometry

Written procedures are established and implemented that address procurement, receipt, inventory, labeling, leak testing, surveillance, control, transfer, disposal, storage, issuance and use of these radioactive sources. These procedures are developed in accordance with the radiation protection program to comply with 10 CFR Parts 19 and 20. A supplementary warning symbol is used in the presence of large sources of ionizing radiation consistent with the guidance in Regulatory Issue Summary (RIS) 2007-03.

Sources maintained on-site for instrument calibration purposes are shielded while in storage to keep personnel exposure ALARA. Sources used to service or calibrate plant instrumentation are also routinely brought on-site by contractors. Radiography is performed by the licensed utility group or licensed contractors. These sources are maintained and used in accordance with the provisions of the utility group's or contractor's license. Additional requirements and restrictions may apply depending on the type of source, use, and intended location of use. If the utility group or contractor source must be stored on-site, designated plant personnel must approve the storage location, and identify appropriate measures for maintaining security and personnel protection.

During the period prior to the implementation of the Emergency Plan (in preparation for the initial fuel loading following the 52.103(g) finding), no specific materials related emergency plan will be necessary because:

a) No byproduct material will be received, possessed, or used in a physical form that is "in unsealed form, on foils or plated sources, or sealed in glass," that exceeds the quantities in Schedule C in 10 CFR 30.72, and

b) No 10 CFR Part 40 specifically licensed source material, including natural uranium, depleted uranium and uranium hexafluoride will be received, possessed, or used during this period.

The following radioactive sources will be used for the Radiation Monitoring System and laboratory/portable monitoring instrumentation:[1]

Radioactive Licensee Material (Element and Mass Number)[1]	Chemical and/or Physical Form[1]	Maximum Quantity That Licensee May Possess at Any One Time[1]
• Any byproduct material with atomic numbers 1 through 93 inclusive	Sealed Sources [2]	No single source to exceed 100 millicuries 5 Curies total
• Americium-241	Sealed Sources [2]	No single source to exceed 300 millicuries 500 millicuries total

1. This information remains in effect between the issuance of the COL and the Commission's 52.103(g) finding for each unit, and will be designated historical information after that time.

2. Includes calibration and reference sources.

12.2.1.3 Sources for the Core Melt Accident

LNP DEP 6.4-1

Revise the last paragraph of DCD Subsection 12.2.1.3 to read as follows:

12.2.1.3.1 Containment

If there is core degradation, core cooling would be provided by the passive core cooling system which is totally inside the containment such that no high activity sump solution would be recirculated outside the containment. The shielding provided for the containment addresses this post-LOCA source term. The source strengths as a function of time are provided in Table 12.2-20 and the integrated source strengths are provided in Table 12.2-21.

12.2.1.3.2 Main Control Room HVAC Filters

During operation of the nuclear island nonradioactive ventilation system (VBS) supplemental filtration or the main control room emergency habitability system (VES), filters in the control room HVAC work to remove particulate and iodine from the air. As radioactivity accumulates within the filters, this becomes a potential source of dose. These source strengths as a function of time are provided in Table 12.2-28 and the integrated source strengths are provided in Table 12.2-29.

12.2.3 COMBINED LICENSE INFORMATION

STD COL 12.2-1 This COL item is addressed in Subsection 12.2.1.1.10.

Table 12.2-201 (Sheet 1 of 2)
Core Melt Accident Source Strengths from MCR HVAC Filters as a Function of Time

LNP DEP 6.4-1

VES Filter [1] Source Strengths after a Loss of Coolant Accident

Energy Group (MeV/gamma)	Source Strength (MeV/sec)			
	2 hours	8 hours	24 hours	30 days
0.01-0.02	1.19E+06	3.11E+06	1.81E+06	1.97E+05
0.02-0.03	1.47E+06	5.26E+06	3.89E+06	2.65E+05
0.03-0.06	2.87E+06	8.30E+06	5.46E+06	6.74E+05
0.06-0.1	3.03E+06	8.13E+06	5.22E+06	5.41E+05
0.1-0.2	5.76E+06	1.41E+07	8.76E+06	9.02E+05
0.2-0.4	6.14E+07	2.61E+08	2.46E+08	1.87E+07
0.4-0.6	1.86E+08	6.02E+08	3.60E+08	1.83E+07
0.6-0.7	1.47E+08	2.33E+08	1.47E+08	1.03E+08
0.7-0.8	1.09E+08	1.80E+08	1.05E+08	7.30E+07
0.8-1.0	1.85E+08	1.67E+08	6.99E+07	7.13E+06
1.0-1.5	3.36E+08	6.99E+08	1.85E+08	1.22E+07
1.5-2.0	1.21E+08	2.55E+08	4.97E+07	2.69E+04
2.0-3.0	3.13E+07	3.87E+07	7.28E+06	9.07E+03
3.0-4.0	3.68E+05	5.98E+03	5.56E+02	1.41E+02
4.0-5.0	1.42E+04	3.16E+01	8.55E-04	7.80E-04
5.0-6.0	3.31E-05	3.12E-04	3.35E-04	3.21E-04
6.0-7.0	1.32E-05	1.24E-04	1.33E-04	1.28E-04
7.0-8.0	5.11E-06	4.82E-05	5.17E-05	4.96E-05
8.0-10.0	2.68E-06	2.53E-05	2.71E-05	2.60E-05
10.0-14.0	1.69E-07	1.60E-06	1.71E-06	1.64E-06
Total	1.19E+09	2.47E+09	1.19E+09	2.35E+08

Notes: 1) Based upon a particulate filter density of 0.212 g/cc and charcoal filter density of 0.440 g/cc.

Table 12.2-201 (Sheet 2 of 2)
Core Melt Accident Source Strengths from MCR HVAC Filters as a Function of Time

LNP DEP 6.4-1

VBS Filter [2] Source Strengths after a Loss of Coolant Accident

Energy Group (MeV/gamma)	Source Strength (MeV/sec) 2 hours	8 hours	24 hours	30 days
0.01-0.02	6.86E+08	1.00E+09	5.75E+08	6.21E+07
0.02-0.03	9.55E+08	1.76E+09	1.27E+09	8.46E+07
0.03-0.06	1.71E+09	2.71E+09	1.75E+09	2.10E+08
0.06-0.1	1.72E+09	2.60E+09	1.63E+09	1.70E+08
0.1-0.2	3.49E+09	4.61E+09	2.81E+09	2.91E+08
0.2-0.4	3.54E+10	8.45E+10	7.59E+10	5.76E+09
0.4-0.6	1.03E+11	1.91E+11	1.10E+11	5.61E+09
0.6-0.7	7.99E+10	7.20E+10	4.39E+10	3.04E+10
0.7-0.8	5.97E+10	5.62E+10	3.17E+10	2.16E+10
0.8-1.0	1.03E+11	5.23E+10	2.13E+10	2.11E+09
1.0-1.5	1.86E+11	2.20E+11	5.64E+10	3.62E+09
1.5-2.0	6.71E+10	8.03E+10	1.53E+10	8.78E+06
2.0-3.0	1.66E+10	1.22E+10	2.24E+09	3.09E+06
3.0-4.0	1.82E+08	1.93E+06	1.89E+05	4.81E+04
4.0-5.0	6.86E+06	7.65E+03	2.91E-01	2.65E-01
5.0-6.0	3.74E-02	1.12E-01	1.14E-01	1.09E-01
6.0-7.0	1.49E-02	4.47E-02	4.54E-02	4.35E-02
7.0-8.0	5.78E-03	1.74E-02	1.76E-02	1.69E-02
8.0-10.0	3.03E-03	9.11E-03	9.24E-03	8.86E-03
10.0-14.0	1.92E-04	5.75E-04	5.84E-04	5.60E-04
Total	6.59E+11	7.82E+11	3.65E+11	7.00E+10

Notes: 2) Based upon a particulate filter density of 0.230 g/cc and charcoal filter density of 0.632 g/cc.

Table 12.2-202
Core Melt Accident Integrated Source Strengths from MCR HVAC Filters

LNP DEP 6.4-1

Energy Group (MeV/gamma)	30-Day Integrated Source Strength (MeV) VES [1]	VBS [2]
0.01-0.02	1.75E+08	5.65E+10
0.02-0.03	3.81E+08	1.26E+11
0.03-0.06	5.89E+08	1.90E+11
0.06-0.1	5.77E+08	1.84E+11
0.1-0.2	9.03E+08	2.95E+11
0.2-0.4	3.34E+10	1.05E+13
0.4-0.6	2.36E+10	7.44E+12
0.6-0.7	3.81E+10	1.15E+13
0.7-0.8	2.63E+10	7.92E+12
0.8-1.0	7.57E+09	2.39E+12
1.0-1.5	1.77E+10	5.67E+12
1.5-2.0	4.03E+09	1.34E+12
2.0-3.0	6.47E+08	2.18E+11
3.0-4.0	1.20E+06	4.46E+08
4.0-5.0	4.17E+04	1.52E+07
5.0-6.0	1.03E-01	3.51E+01
6.0-7.0	4.08E-02	1.40E+01
7.0-8.0	1.59E-02	5.42E+00
8.0-10.0	8.32E-03	2.84E+00
10.0-14.0	5.25E-04	1.80E-01
Total	1.54E+11	4.79E+13

Notes: 1) Based upon a particulate filter density of 0.212 g/cc and charcoal filter density of 0.440 g/cc.
2) Based upon a particulate filter density of 0.230 g/cc and charcoal filter density of 0.632 g/cc.

12.3 RADIATION PROTECTION DESIGN FEATURES

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

12.3.2.2.7 Control Room Shielding Design

LNP DEP 6.4-1

Revise DCD Subsection 12.3.2.2.7 to read as follows:

The design basis loss-of-coolant accident dictates the shielding requirements for the control room. The rod ejection accident dictates the shielding requirements for the main control room emergency habitability (VES) filter in the operator break room. Consideration is given to shielding provided by the shield building structure. Shielding combined with other engineered safety features is provided to permit access and occupancy of the control room following a postulated loss-of-coolant accident, so that radiation doses are limited to five rem whole body from contributing modes of exposure for the duration of the accident, in accordance with General Design Criterion 19.

Shielding of the VES filtration unit is accomplished by safety-related metal shielding. This shielding is composed of either tungsten that is 0.25 inches thick or stainless steel shown to provide an equivalent amount of shielding. The length and width of the shielding are designed to match the length and width of the filtration unit being shielded.

12.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

Add the following text to the end of DCD Subsection 12.3.4.

STD COL 12.3-2

Procedures detail the criteria and methods for obtaining representative measurement of radiological conditions, including in-plant airborne radioactivity concentrations in accordance with applicable portions of 10 CFR Part 20 and consistent with the guidance in Regulatory Guides 1.21-Appendix A, 8.2, 8.8, and 8.10. Additional discussion of radiological surveillance practices is included in the radiation protection program description provided in Appendix 12AA.

Surveillance requirements are determined by the functional manager in charge of radiation protection based on actual or potential radiological conditions encountered by personnel and the need to identify and control radiation, contamination, and airborne radioactivity. These requirements are consistent with the operational philosophy in Regulatory Guide 8.10. Frequency of scheduled surveillance may be altered by permission of the functional manager in charge of radiation protection or their designee. Radiation Protection periodically provides

cognizant personnel with survey data that identifies radiation exposure gradients in area resulting from identified components. This data includes recent reports, with survey data, location and component information.

The following are typical criteria for frequencies and types of surveys:

Job Coverage Surveys

- Radiation, contamination, and/or airborne surveys are performed and documented to support job coverage.
- Radiation surveys are sufficient in detail for Radiation Protection to assess the radiological hazards associated with the work area and the intended/specified work scope.
- Surveys are performed commensurate with radiological hazard, nature and location of work being conducted.
- Job coverage activities may require surveys to be conducted on a daily basis where conditions are likely to change.

Radiation Surveys

- Radiation surveys are performed at least monthly in any radiological controlled area (RCA) where personnel may frequently work or enter. Survey frequencies may be modified by the functional manager in charge of radiation protection as previously noted.
- Radiation surveys are performed prior to or during entry into known or suspected high radiation areas for which up to date survey data does not exist.
- Radiation surveys are performed prior to work involving highly contaminated or activated materials or equipment.
- Radiation surveys are performed at least semiannually in areas outside the RCA. Areas to be considered include shops, offices, and storage areas.
- Radiation surveys are performed to support movement of highly radioactive material.
- Neutron radiation surveys are performed when personnel may be exposed to neutron emitting sources.

Contamination Surveys

- Contamination surveys are performed at least monthly in any RCA where personnel may frequently work or enter. Survey frequencies may be modified by the functional manager in charge of radiation protection as previously noted.

- Contamination surveys are performed during initial entry into known or suspected contamination area(s) for which up to date survey data does not exist.

- Contamination surveys are performed at least daily at access points, change areas, and high traffic walkways in RCAs that contain contaminated areas. Area access points to a High Radiation Area or Very High Radiation Area are surveyed prior to or upon access by plant personnel or if access has occurred.

- Contamination surveys are performed at least semiannually in areas outside the RCA. Areas to be considered include shops, offices, and storage areas.

- A routine surveillance is conducted in areas designated by the functional manager in charge of radiation protection or their designee likely to indicate alpha radioactivity. If alpha contamination is identified, frequency and scope of the routine surveillance is increased.

Airborne Radioactivity Surveys

- Airborne radioactivity surveys are performed during any work or operation in the RCA known or suspected to cause airborne radioactivity (e.g., grinding, welding, burning, cutting, hydrolazing, vacuuming, sweeping, use of compressed air, using volatiles on contaminated material, waste processing, or insulation).

- Airborne radioactivity surveys are performed during a breach of a radioactive system, which contains or is suspected of containing significant levels of contamination.

- Airborne radioactivity surveys are performed during initial entry (and periodically thereafter) into any known or suspected airborne radioactivity area.

- Airborne radioactivity surveys are performed immediately following the discovery of a significant radioactive spill or spread of radioactive contamination, as determined by the functional manager in charge of radiation protection.

- Airborne radioactivity surveys are performed daily in occupied radiological controlled areas where the potential for airborne radioactivity exists, including containment.

- Airborne radioactivity surveys are performed any time respiratory protection devices, alternative tracking methods such as derived air concentration-hour (DAC-hr), and/or engineering controls are used to control internal exposure.

- Airborne radioactivity surveys are performed using continuous air monitors (CAMs) for situations in which airborne radioactivity levels can fluctuate and early detection of airborne radioactivity could prevent or minimize inhalations of radioactivity by workers. Determination of air flow patterns are considered for locating air samplers.

- Airborne radioactivity surveys are performed prior to use and monthly during use on plant service air systems used to supply air for respiratory protection to verify the air is free of radioactivity.

- Tritium sampling is performed near the spent fuel pit when irradiated fuel is in the pit and other areas of the plant where primary system leaks occur and tritium is suspected.

Appropriate counting equipment is used based on the sample type and the suspected identity of the radionuclides for which the sample is being done. Survey results are documented, retrievable, and processed per site document control and records requirements consistent with Regulatory Guide 8.2. Completion of survey documentation includes the update of room/area posting maps and revising area or room postings and barricades as needed.

Air samples indicating activity levels greater than a procedure specified percentage of DAC are forwarded to the radiochemistry laboratory for isotopic analysis. Samples which cannot be analyzed on-site are forwarded to an offsite laboratory or a contractor for analysis; or, the DAC percentage may be hand calculated using appropriate values from 10 CFR Part 20, Appendix B.

The responsible radiation protection personnel review survey documentation to evaluate if surveys are appropriate and obtained when required, records are complete and accurate, and adverse trends are identified and addressed.

An in-plant radiation monitoring program maintains the capability to accurately determine the airborne iodine concentration in areas within the facility where personnel may be present under accident conditions. This program includes the training of personnel, procedures for monitoring, and provisions for maintenance of sampling and analysis equipment consistent with Regulatory Guides 1.21 (Appendix A) and 8.8. Training and personnel qualifications are discussed in Appendix 12AA.

A portable monitor system meeting the requirements of NUREG-0737, Item III.D.3.3, is available. The system uses a silver zeolite or charcoal iodine sample cartridge and a single-channel analyzer. The use of this portable monitor is incorporated in the emergency plan implementing procedures. The portable monitor is part of the in-plant radiation monitoring program. It is used to determine the airborne iodine concentration in areas where plant personnel may be present during an accident. Accident monitoring instrumentation complies with applicable parts of 10 CFR Part 50, Appendix A.

Sampling cartridges can be removed to a low background area for further analysis. These cartridge samples can be purged of any entrapped noble gases, when necessary, prior to being analyzed.

12.3.5.1 Administrative Controls for Radiological Protection

STD COL 12.3-1 This COL Item is addressed in Subsection 12.5.4 and Appendix 12AA.

12.3.5.2 Criteria and Methods for Radiological Protection

STD COL 12.3-2 This COL Item is addressed in Subsection 12.3.4.

12.3.5.3 Groundwater Monitoring Program

STD COL 12.3-3 This COL Item is addressed in Appendix 12AA.

12.3.5.4 Record of Operational Events of Interest for Decommissioning

STD COL 12.3-4 This COL Item is addressed in Appendix 12AA.

12.4 DOSE ASSESSMENT

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

LNP SUP 12.4-1

Add the following new subsection after DCD Subsection 12.4.1.8:

12.4.1.9 Dose to Construction Workers

This section evaluates the potential radiological dose impacts to construction workers at the Levy Nuclear Plant, Units 1 and 2 (LNP). Since a portion of the LNP 2 construction period overlaps operation of LNP 1, construction workers at LNP 2 would be exposed to direct radiation and gaseous radioactive effluents from LNP 1. Doses to construction workers during construction of LNP 1 are not evaluated since the only radiation sources prior to the start-up of LNP 1 are background sources.

12.4.1.9.1 Site Layout

Figure 2.1.1-203 indicates the locations of LNP 1 and 2. Construction activity for LNP 2 is outside the protected area for LNP 1 but inside the restricted area boundary.

12.4.1.9.2 Radiation Sources

Duke Energy Florida, LLC (DEF) proposes to construct LNP 1 first. Construction workers at a new facility on the LNP site are not exposed to any radiation sources until LNP 1 becomes operational. Radiation dose to construction workers on LNP 2 is from direct radiation from LNP 1, from airborne effluents from LNP 1, and from background radiation.

The radiation exposure at the site boundary is considered in DCD Section 12.4.2. As stated in that section, direct radiation from the containment and other plant buildings is negligible. Additionally, there is no contribution from refueling water since the refueling water is stored inside the containment instead of in an outside storage tank.

Small quantities of monitored airborne effluents are normally released through the plant vent or the turbine building vent. The plant vent provides the release path for containment venting releases, auxiliary building ventilation releases, annex building releases, radwaste building releases, and gaseous radwaste system discharge. The turbine building vents provide the release path for the condenser air removal system, gland seal condenser exhaust, and the turbine building ventilation releases. The ventilation system is described in DCD Section 9.4. The expected radiation sources (nuclides and activities) in the gaseous effluents are listed in DCD Table 11.3-3.

Exposure of LNP 2 construction workers to radioactive liquid effluents was not evaluated because the discharge structure and blowdown piping will be completed during LNP 1 construction.

12.4.1.9.3 Construction Worker Dose Estimates

The determination of construction worker doses from LNP 1 operation depends on the airborne effluent released and the atmospheric transport to the worker location. The methodology contained in the GASPAR II program was used to determine the doses for gaseous pathways. This program implements the radiological exposure models described in Regulatory Guide 1.109 for radioactivity releases in gaseous effluent.

Dose rate estimates were calculated for construction workers exposed to gaseous radioactive effluents through the following pathways:

- Direct radiation from immersion in the gaseous effluent plume and from particulates deposited on the ground.

- Inhalation of gases and particulates.

For the purposes of these calculations, the Chi/Q value (1.52E-04 seconds per cubic meter [sec/m^3] calculated at 402 meters (m) (1320 feet [ft.]) in the worst meteorological sector (WSW), obtained from Table 2.3.5-201, was used in GASPAR II to calculate construction worker doses from the gaseous pathway, and conservatively bounds the construction worker location at LNP 2. LNP 2 will be situated directly due north of LNP 1 (in the N sector), where the Chi/Q value at 402 m (1320 ft.) is 2.20E-05 sec/m^3. Use of the WSW sector Chi/Q provides more than a factor of 6 margin in the calculated construction worker doses.

GASPAR II doses calculated at 402 m (1320 ft.) were adjusted based on construction worker residence time on the site or 2080 hours/8,760 hours = 0.24. Results are presented in the following subsections.

12.4.1.9.4 Compliance with Dose Regulations

LNP 2 construction workers are, for the purposes of radiation protection, members of the general public. This means that the dose rate limit (TEDE) is 1.0 milliSievert (mSv) (100 millirem per year [mrem/year]). The construction workers (with the exception of certain specialty contractors loading fuel or using industrial radiation sources for radiography) do not deal with radiation sources.

There are three regulations that govern dose rates to members of the general public. Dose rate limits to the public are provided in 10 Code of Federal Regulations (CFR) 20.1301, 10 CFR 20.1302, and 10 CFR Part 50, Appendix I. The design objectives of 10 CFR 50, Appendix I, apply relative to maintaining dose as ALARA for construction workers. In addition, 40 CFR Part 190 applies as it is referred to in 10 CFR 20.1301. The requirements of 10 CFR 20.1201 through

20.1204 do not apply to the construction workers because they are considered members of the public and not radiation workers.

12.4.1.9.5 Collective Doses to LNP Unit 2 Workers

Collective construction worker doses were conservatively estimated using the following information:

- The estimated maximum dose rate (TEDE) for the gaseous pathway.
- A construction worker exposure time of 2,080 hours per year (40 hours per week for 52 weeks).
- An estimated peak loading of 2,700 construction workers per year for LNP 2 construction.

The collective dose is the sum of all doses received by all workers. It is a measure of population risk. The total annual worker collective dose is 0.119 person-Sievert (person-Sv) (11.9 person-rem) from LNP 1 operation (excluding background radiation). The contribution to the annual average dose received by an individual from background radiation as reported in Table 5.7-A-2 of NUREG-1555 is approximately 3.6 mSv/yr (360 mrem/yr). Multiplying the annual average dose received by the construction workforce assumed yields a collective dose of 9.72 person-Sv from background and man-made radiation. Doses to construction workers from LNP 1 operations are less than 1.5 percent of those received from background radiation.

Table 12.4-201 compares the estimated doses (TEDE) to a LNP construction worker with the public dose criteria of 10 CFR 20.1301. This comparison demonstrates compliance with 10 CFR 20.1301 criteria and supports the conclusion that those who will construct the LNP facility would not need to be classified as radiation workers nor would they require monitoring.

It has been assumed that a construction worker exposure time is 40 hours per week for 52 weeks. Since construction projects can involve significant overtime for construction workers the annual dose to a construction worker could be higher. The dose would be higher in proportion to the amount of time over 40 hours per week. If the construction workers worked 60 hours per week for 52 weeks, the dose presented in Table 12.4-201 would increase by 50%. However, when compared to the dose limits presented in Table 12.4-201, the dose to the construction worker is still well below the 10 CFR 20.1301 public dose limit. This would be the case for any expected level of overtime including the most limiting 72 hour or 84 hour work weeks.

12.4.1.9.6 Operating Unit Radiological Surveys

STD SUP 12.4-1

The operating unit conducts radiological surveys in the unrestricted and controlled area and radiological surveys for radioactive materials in effluents

discharged to unrestricted and controlled areas in implementing 10 CFR 20.1302. These surveys demonstrate compliance with the dose limits of 10 CFR 20.1301 for construction workers.

LNP SUP 12.4-1

Table 12.4-201
Comparison of LNP Construction Worker Estimated Radiation Doses to 10 CFR 20.1301 Public Dose Criteria

Type of Radiation Dose	Public Dose Limits 10 CFR 20.1301	Estimated LNP Construction Worker Dose
Total effective dose equivalent (TEDE)	100 mrem/yr	0.044 mSv/yr (4.4 mrem/yr)
Maximum dose in any one hour	2 mrem	2.1E-05 mSv (2.1E-03 mrem)

12.5 HEALTH PHYSICS FACILITIES DESIGN

This section of the referenced DCD is incorporated by reference with the following departures and/or supplements.

12.5.4 CONTROLLING ACCESS AND STAY TIME

STD COL 12.3-1

Add the following text to the end of DCD Subsection 12.5.4.

A closed circuit television system may be installed in high radiation areas to allow remote monitoring of individuals entering high radiation areas by personnel qualified in radiation protection procedures.

12.5.5 COMBINED LICENSE INFORMATION

STD COL 12.5-1

This COL Item is addressed in Appendix 12AA.

Add the following Appendix after Section 12.5 of the DCD.

APPENDIX 12AA RADIATION PROTECTION PROGRAM DESCRIPTION

STD COL 12.1-1
STD COL 12.3-1
STD COL 12.5-1

This appendix incorporates by reference NEI 07-03A, Generic FSAR Template Guidance for Radiation Protection Program Description. See Table 1.6-201. The numbering of NEI 07-03A is revised from 12.5# to 12AA.5# through the document, with the following revisions and additions as indicated by strikethroughs and underlines. Table 13.4-201 provides milestones for radiation protection program implementation.

Revise bullet number 3 of NEI 07-03A Section 12.5 as follows:

3. Prior to initial loading of fuel in the reactor, all of the radiation program functional areas described in Appendix 12AA~~Section 12.5~~ will be fully implemented, with the exception of the organization, facilities, equipment, instrumentation, and procedures necessary for transferring, transporting or disposing of radioactive materials in accordance with 10 CFR Part 20, Subpart K, and applicable requirements in 10 CFR Part 71. In addition, the position of radiation protection manager (as described in ~~s~~Section 13.1 ~~12.5.2.3~~) will be filled and at least one (1) radiation protection technician for each operating shift, selected, trained, and qualified consistent with the guidance in Regulatory Guide 1.8, will be onsite and on duty when fuel is initially loaded in the reactor, and thereafter, whenever fuel is in the reactor.

Revise the first paragraph of NEI 07-03A Subsection 12.5.2 as follows:

Qualification and training criteria for site personnel are consistent with the guidance in Regulatory Guide 1.8 and are described in FSAR Chapter 13. Specific radiation protection responsibilities for key positions within the plant organization are described in Section 13.1 ~~below~~.

Subsections 12.5.2.1 through 12.5.2.5 of NEI 07-03A are not incorporated into Appendix 12AA.

Subsection 12.5.3.1 of NEI 07-03A is not incorporated into Appendix 12AA. Facilities are described in DCD Subsection 12.5.2.2.

Add the following text after the first paragraph of NEI 07-03A Subsection 12.5.3.3.

If circumstances arise in which NIOSH tested and certified respiratory equipment is not used, compliance with 10 CFR 20.1703(b) and 20.1705 is maintained.

The following headings (and associated material) in Subsection 12.5.4.2 of NEI 07-03A are described in DCD Subsection 12.5.3, and are therefore not incorporated into Appendix 12AA:

- Radwaste Handling
- Spent Fuel Handling
- Normal Operation
- Sampling

Add the following text after the second paragraph of NEI 07-03A Subsection 12.5.4.4.

STD COL 12.3-1

Table 12AA-201 identifies plant areas designated as Very High Radiation Areas (VHRAs), lists corresponding plant layout drawings showing the VHRA in DCD Section 12.3, specifies the condition under which the area is designated VHRA, identifies the primary source of the VHRA, and summarizes the frequency of access and reason for access. VHRAs are listed as Radiation Zone IX, which corresponds to a dose rate greater than 500 rad/hr.

In each of the VHRAs, with the exception of the Reactor Vessel Cavity and Delay-Bed / Guard-Bed Compartment, the primary radioactive source is transient (such as fuel passing through the transfer tube), removable (such as resin in the demineralizers), or can be relocated. When the primary source is removed, the dose rate in each of these areas will be less than Zone IX and, in effect, the area will no longer be a VHRA. With planning, the need for human entrance to a VHRA when the primary source is present can be largely or entirely avoided.

In addition to the access control requirements for high radiation areas, the following control measures are implemented to control access to very high radiation areas in which radiation levels could be encountered at 500 rads or more in one hour at one meter from a radiation source or any surface through which the radiation penetrates:

- Sign(s) conspicuously posted stating GRAVE DANGER, VERY HIGH RADIATION AREA.
- Area is locked. Each lock shall have a unique core. The keys shall be administratively controlled by the functional manager in charge of radiation protection as described in Section 13.1.
- Plant Manager's (or designee) approval required for entry.
- Radiation Protection personnel shall accompany person(s) making the entry. Radiation Protection personnel shall assess the radiation exposure conditions at the time of the entry.

A verification walk down will be performed with the purpose of verifying barriers to the Very High Radiation Areas in the final design of the facility are consistent with Regulatory Guide 8.38 guidance as part of the implementation of the Radiation Protection and ALARA programs on the schedule identified in Table 13.4-201.

Revise the third paragraph of NEI 07-03A Subsection 12.5.4.7 as follows.

STD COL 12.1-1
STD COL 12.3-1
STD COL 12.5-1

As described in Sections 12.1, ~~12.5.1~~Appendix 12AA and ~~12.5.2~~ 13.1, management policy is established, and organizational responsibilities and authorities are assigned to implement an effective program for maintaining occupational radiation exposures ALARA. Procedures are established and implemented that are in accordance with 10 CFR 20.1101 and consistent with the guidance in Regulatory Guides 8.8 and 8.10. Examples of such procedures include the following:

Add the following text after the last bullet of NEI 07-03A Subsection 12.5.4.8.

STD COL 12.5-1

This subsection adopts NEI 08-08A (Reference 201), for a description of the operational and programmatic elements and controls that minimize contamination of the facility, site, and the environment, to meet the requirements of 10 CFR 20.1406.

Revise the first paragraph of Subsection 12.5.4.12 of NEI 07-03A to read:

STD COL 12.5-1

The radiation protection program and procedures are established, implemented, maintained, and reviewed consistent with the 10 CFR 20.1101 and the quality assurance criteria described in Part III of the Quality Assurance Program Description described in Section 17.5.

Add the following Subsection to the information incorporated from NEI 07-03A.

STD COL 12.3-3

12AA.5.4.14 Groundwater Monitoring Program

A groundwater monitoring program beyond the normal radioactive effluent monitoring program is developed. If necessary to support this groundwater monitoring program, design features will be installed during the plant construction process. Areas of the site to be specifically considered in this groundwater monitoring program are (all directions based on plant standard):

- West of the auxiliary building in the area of the fuel transfer canal.
- West and south of the radwaste building.

- East of the auxiliary building rail bay and the radwaste building truck doors

This subsection adopts NEI 08-08A (Reference 201), for the Groundwater Monitoring Program description.

Add the following Subsection to the information incorporated from NEI 07-03A.

STD COL 12.3-4

12AA.5.4.15 Record of Operational Events of Interest for Decommissioning

This subsection adopts NEI 08-08A (Reference 201), for discussion of record keeping practices important to decomissioning.

Revise the REFERENCES section of NEI 07-03A, Reference 8, to read as follows:

8. Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."~~4, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants."~~

Add the following reference to NEI 07-03A REFERENCES.

201. NEI 08-08A, Generic FSAR Template Guidance for Life Cycle Minimization of Contamination, Revision 0, October 2009 (ML093220445).

STD COL 12.3-1

Table 12AA-201 (Sheet 1 of 2)
Very High Radiation Areas (VHRA)

Room Number	VHRA Location	DCD Figure 12.3-1, Sheet No.	Primary Source(s)	VHRA Conditional Notes	Frequency to Access of VHRA Areas While VHRA Conditions Exist
11105	Reactor Vessel Cavity	3, 4, 5	Neutron activation of the material in and around the cavity during reactor operations, such as the concrete shield walls and the reactor insulation	Note 1	None Required
12151	Spent Fuel Pool Cooling System / Liquid Radwaste System Demineralizer / Filter room (Inside Wall)	3	Resin in vessels	Notes 6, 8	None Required
12153	Delay-Bed / Guard-Bed Compartment	3	Activated carbon holding radioactive gases	Note 10	None Required
12371	Filter-Storage Area	6, 7	Spent filter cartridges	Notes 4, 6, 7	None Required
12372	Resin Transfer Pump / Valve Room	6	Spent resin in lines	Note 6	None Required
12373	Spent-Resin Tank Room	6	Spent resin in tanks	Note 6	None Required
12374	Waste Disposal Container Area	6	Spent resin in vault	Note 6	None Required
12463	Cask Loading Pit	6	Spent fuel	Notes 2, 6	None Required
12563	Spent Fuel Pit	5, 6	Spent fuel	Note 6	None Required
Fuel Transfer Areas					
12564	Fuel Transfer Tube	6	Fuel in transit	Notes 2, 5, 9	None Required
11205	Reactor Vessel Nozzle Area	5	Fuel in transit	Notes 2, 3, 9	None Required
11504	Refueling Cavity	6	Fuel in transit	Notes 2, 3, 9	None Required

STD COL 12.3-1

Table 12AA-201 (Sheet 2 of 2)
Very High Radiation Areas (VHRA)

Notes

1. VHRA during full power operation; less than 10 Rem/hr 24 hours after plant shutdown.
2. During underwater spent fuel transfer operations, this area can be as high as VHRA.
3. During underwater reactor internals transfers / storage, this area can be as high as VHRA.
4. During spent resin waste disposal container transfer or loading, this area can be as high as VHRA. The contact dose rate of spent resin containers can be greater than 1000 Rem/hr.
5. Discussion about the Spent Fuel Transfer Canal and Tube Shielding is provided in DCD Subsection 12.3.2.2.9.
6. Source is transient, removable, or can be relocated.
7. VHRA when hatch is removed during spent resin container handling operation.
8. In the event that the room does need to be accessed for maintenance or other reasons, temporary shielding is put in place and the resin is removed from the vessels. These measures reduce exposure rates in the room, such that this room is no longer VHRA. Remote handling is used for any tasks that require the opening of the access hatch in the ceiling of this room when media is present.
9. These areas have no planned reasons for entry and are only classified as VHRAs during periods of fuel movement. In the event that these rooms do need to be accessed to repair the Fuel-Transfer System, Fuel Transfer Tube Gate Valve, or other components, it is done during a non-fuel movement time. This keeps the dose received by the worker as low as reasonably achievable.
10. Inspection of the equipment in this room, when required, is done using remote viewing equipment. Two plugs between Room 12153 and 12155 contain instruments and the plugs are expected to be removed every 12 to 18 months for performance of maintenance. Administrative procedures are implemented to protect workers pursuant to Regulatory Guide 8.38.