CHAPTER 12

RADIATION PROTECTION

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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

Notes:

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.

12.2.3 COMBINED LICENSE INFORMATION

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

12.3 RADIATION PROTECTION DESIGN FEATURES

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

12.3.4 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:

<u>Job Coverage Surveys</u>

- 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.

HAR 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 assesses the potential radiological dose impacts to those who will construct the proposed Shearon Harris Nuclear Power Plant Units 2 and 3 (HAR) and be exposed to the existing Shearon Harris Nuclear Power Plant, Unit 1 (HNP) and HAR 2 during construction of HAR 3.

12.4.1.9.1 Site Layout

Figure 2.1.1-203 indicates the locations of HAR 2 and 3 relative to the layout of various HNP facilities. HAR 2 is located approximately 427 m (1400 feet) north and west of the HNP containment. HAR 3 is located approximately 290 m (950 feet) north and west of HAR 2 and approximately 716 m (2350 feet) north and west of the HNP containment. The major activities during the construction of HAR 2 and 3 are expected to take place outside the HNP protected area boundary, but inside the restricted area boundary.

12.4.1.9.2 Radiation Sources

HAR 2 and 3 construction workers could be exposed to any elevated background levels and effluent discharges from current HNP reactor operations. Once HAR 2 is operational, workers involved with the construction of HAR 3 could be exposed to radiation sources from HAR 2.

Total dose exposure comparison for HAR 2 and HAR 3 determined that HAR 3 construction worker exposure was the most limiting and thus bounds HAR 2 construction worker exposure.

In determining the dose to HAR 3 construction workers, the following radiation sources are considered:

- Direct radiation exposure from HNP.
- Direct radiation exposure from HAR 2.
- HNP gaseous effluent releases.
- HAR 2 gaseous effluent releases.
- HNP and HAR 2 liquid effluent releases to Harris Lake that is the source of drinking water for HAR 2 and 3 construction workers.

Exposure of HAR 3 workers to radioactive liquid effluents due to shared systems between HAR 2 and HAR 3 was not evaluated because the discharge structure and blowdown piping for HAR 3 will be completed during HAR 2 construction.

12.4.1.9.3 Measured Radiation Dose Rates and Liquid/Airborne Concentrations

During construction of the HAR 2 facility, construction workers may be exposed to direct radiation and to the radioactive effluents emanating from the routine operation of the HNP. During construction of the HAR 3 facility, construction workers may be exposed to direct radiation from HAR 2 and to the radioactive effluents emanating from the routine operation of the HNP and HAR 2.

Total dose exposure comparison for HAR 2 and HAR 3 determined that HAR 3 construction worker exposure was the most limiting and thus bounds HAR 2 construction worker exposure.

12.4.1.9.3.1 Liquid Effluent Doses

12.4.1.9.3.1.1 HNP Liquid Effluent Doses

Radioactive liquids are routinely released as batches from the waste evaporator condensate tank and the treated laundry and hot shower tank. Batch releases may also originate from the secondary waste sample tank and the waste monitor tank at the HNP. Based on analysis of the tank contents, the tank release rate is adjusted, based on the cooling tower blowdown line flow rate, to dilute the tank activities to 50 percent of the allowable concentrations at the release point to Harris Lake (Reference 202). The liquid effluent release point is at the point of discharge from the cooling tower blowdown line into Harris Lake. The cooling tower blowdown line provides liquid effluent dilution prior to release to Harris Lake. Concurrent batch releases do not occur at the HNP. The secondary waste sample tank and the normal service water system have a low potential for radioactive effluent releases. Effluent monitors on the secondary waste sample tank and the normal service water lines check these releases (Reference 202).

Since it is unlikely that the HAR construction workers will be exposed to liquid effluent pathways, it is assumed that the liquid effluent dose rates to which the workers will be exposed are the same as those for the maximally exposed member of the public. The estimated maximum individual off-site doses due to radioactivity released in the HNP's liquid effluent release pathway for the period from 1999 through 2010 was 1.86 E-02 mrem per year (mrem/yr) total body and 2.63 E-02 mrem/yr maximum organ in 2004; and 1.99 E-02 mrem/yr total body and 1.99 E-02 mrem/yr maximum organ in 2007 (Reference 201 and Reference 205). This dose is considered to be a negligible contributor to a HAR 2 or HAR 3 construction worker total dose. The annual releases for 2004 and 2007 were selected because they resulted in the maximum exposure to the public among the years 1999-2010 (Reference 201 and Reference 205).

Liquid effluent dose associated with tritium released into the Harris Lake is applicable to HAR 2 and 3 construction workers. During the period of January 1, 2008 through December 31, 2008 (Reference 203), the tritium dose from drinking water obtained from Harris Lake to the worker at the Wake County Fire Training Center was equal to 1.24 E-01 mrem. Construction workers are assumed to use Harris Lake as a drinking water source. The exposure to construction workers due to liquid effluents discharged by HNP into Harris Lake is considered to be equivalent to the exposure of the worker at the Wake County Fire Training Center in 2008. The January 1, 2008 through December 31, 2008 period was chosen because it identifies the maximum tritium drinking water dose since the drinking water pathway was identified for inclusion in the total dose determination.

12.4.1.9.3.1.2 HAR 2 Liquid Effluent Doses

In accordance with plant procedures, small amounts of liquid radioactive effluents (below regulatory limits) will be mixed with the cooling water and discharged to Harris Lake. Construction workers are assumed to use Harris Lake as a drinking water source. The LADTAP II computer program was used to calculate the construction worker doses from the liquid pathway via the ingestion of drinking water from Harris Lake. Calculations resulted in a whole body dose of 0.7 mrem per year (mrem/yr). PEC maintains USEPA drinking water standards for water taken from Harris Lake for use as drinking water at the Harris Site.

12.4.1.9.3.2 Gaseous Effluent Releases

12.4.1.9.3.2.1 HNP Gaseous Effluent Releases

At HNP, four gaseous effluent discharge points exist: Plant Vent Stack 1, Turbine Building Vent Stack 3A, and the Waste Processing Building Vent Stacks 5 and 5A. During refueling outages, when the equipment hatch is removed, there is the potential for airborne particulate releases. All gaseous effluent releases at the plant are considered ground releases (Reference 202).

If the reactor has been shut down for greater than 30 days, the condenser vacuum pump discharge during initial hogging operations at plant start-up and prior to turbine operation may be routed as dual exhaust to (1) the Turbine Building Vent Stack 3A and (2) the atmosphere directly (Reference 202).

The stack effluent monitor setpoints ensure that the dose rates from noble gases at the HAR site boundary do not exceed the applicable regulatory limits established for releases to unrestricted areas (Reference 202). Data from the HNP Radioactivity Effluent Release Report for the period of January 1, 2010 through December 31, 2010 provided the maximum doses to the public for the 1999-2010 period (Reference 206).

HNP gaseous effluent release exposure was determined by using the HNP Annual Effluent Release Report data for gaseous effluents and the calculation of

the dose at the Exclusion Area Boundary using the methodology presented in Subsection 3.3.1 of the HNP ODCM (Reference 202). The methodology was then adjusted for the Chi/Q of the worst meteorological sector and the location of HAR 3 construction workers. The dose from HNP gaseous effluent releases at the location of HAR 3 construction workers (716 m or 2350 ft. from HNP) is 14.6 mrem/yr. When adjusted for construction worker residence time on the site (2080 hours/8760 hours = 0.24) the dose to a construction worker located at HAR 3 from HNP is 3.5 mrem/yr.

12.4.1.9.3.2.2 HAR 2 Gaseous Effluent Releases

The determination of construction worker doses from HAR 2 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.

HAR 2 gaseous effluent release exposure was based on the dose from the gaseous effluent release at the Exclusion Area Boundary as determined by the GASPAR II program. The GASPAR II methodology was then used to adjust the dose using the Chi/Q at the worst meteorological sector (SSW) and the HAR 3 construction worker location of 290 meters (m) (950 feet). The HAR 3 construction worker dose was calculated to be 11.9 mrem/yr. When adjusted for construction worker residence time on the site per year (2080 hours/8760 hours = 0.24) the dose to a construction worker located at HAR 3 from HAR 2 is 2.9 mrem/yr.

12.4.1.9.3.3 Direct Radiation Measurements

12.4.1.9.3.3.1 Direct Radiation Exposure from HNP

Direct radiation exposure input was determined from HNP protected area fence line Thermo Luminescent Dosimeter (TLD) readings that have been compiled over approximately seven years, from the first quarter of 1999 through the third quarter of 2006 (Reference 204). There are 16 TLD locations along the HNP protected area fence line as shown on Figure 12.4-201.

Selecting the individual TLD dose data for the TLDs that are closest to the HAR 2 construction boundary (TLDs 6, 7 and 8 per Figure 12.4-201), identifies the highest peak dose to be approximately 32 mrem for any 90-day period (from TLD

7) (Reference 204). TLDs 15 and 24 have peak dose rates higher than TLD # 7; however, TLD # 15 is located on the opposite end of HNP with respect to the HAR 2 construction site and TLD # 24 is located on the 4th floor of K Building. Neither TLD # 15 nor TLD # 24 would provide dose measurements representative of construction worker doses.

The maximum dose of gamma radiation over any 90-day period for TLD # 7 was approximately 32 mrem (without background correction) as shown on Figure 12.4-202. Using the 32 mrem per 90-day period value for TLD # 7 for estimating the doses to construction workers is considered both reasonable and conservative because:

- The HAR facilities will be located outside the HNP protected area fence line and will be away from any HNP radiation sources. The HNP TLD locations that are the closest to HAR 2 are TLD # 6, 7, and 8. These TLDs are over 300 feet from the closest HAR 2 structures.

- TLD # 7 has the highest peak dose over any 90-day period of the TLDs located closest to the construction workers.

- The majority of the construction workers will be located in the HAR 2 and HAR 3 nuclear and turbine island construction areas which are further from the HNP operating radiation sources than the distances identified in the closest HAR 2 structures reflected in the protected area fence line TLD #6, 7, and 8 locations.

- No credit for the reduction in potential dose is given for the distance from the HNP protected area fence line TLD locations to the HAR facility construction areas or the potential shielding effects HAR 2 would provide to construction workers at HAR 3.

The direct radiation exposure for construction workers was based on a 2,080-hour work year and an exposure rate of 14.8 μrem/hr or 31 mrem/yr.

12.4.1.9.3.3.2 Direct Radiation Exposure from HAR 2

The dose calculated for HAR 2 construction workers does not take credit for the reduction in potential dose rate due to the separation distance of the plants. AP1000 DCD Subsection 12.3.2.2.1 states, “During reactor operation, the shield building protects personnel occupying adjacent plant structures and yard areas from radiation originating in the reactor vessel and primary loop components. The concrete shield building wall and the reactor vessel and steam generator compartment shield walls reduce radiation levels outside the shield building to less than 0.25 mrem/hr from sources inside containment. The shield building completely surrounds the reactor coolant system components.” With a design basis dose rate directly outside the HAR 2 shield building of less than 0.25 mrem/hr, the dose at the fence around the HAR 2 protected area would be negligible. Thus, the contribution to construction workers from the HAR 2

containment and other buildings would be negligible. Therefore, the doses to HAR 3 construction workers from active HAR 2 operations would be negligible.

12.4.1.9.4 Construction Worker Dose Estimates

Annual potential radiological dose impacts to construction workers have been conservatively estimated based on the following factors:

- Total dose exposure comparison for HAR 2 and HAR 3 determined that HAR 3 construction worker exposure was the most limiting and bounds HAR 2 construction worker exposure.
- The liquid effluent release dose (other than tritium) is considered to be a negligible contributor to a HAR 2 or 3 construction worker total dose.
- The estimated maximum construction worker on-site dose due to the drinking water pathway from HNP liquid effluent releases to Harris Lake was 1.24 E-01 mrem/yr (Reference 203).
- The estimated maximum construction worker on-site dose due to the drinking water pathway from HAR 2 liquid effluent releases to Harris Lake was 0.7 mrem/yr.
- Thus, the total estimated construction worker dose due to the drinking water exposure pathway for liquid effluents from both HNP and HAR 2 is approximately 0.8 mrem/yr.
- The estimated radiological exposure to a construction worker from the operation of the HNP via the gaseous effluent release pathway is 3.5 mrem/yr. This is based on the HNP annual effluent release report for gaseous effluents and the calculation of the dose at the exclusion area boundary (EAB) using the methodology of the Off-Site Dose Calculation Manual (ODCM) and then adjusting the dose for the Chi/Q in the worst meteorological sector and the location of the construction workers at HAR 3.
- The estimated radiological exposure to a construction worker from the operation of the HAR via the gaseous effluent release pathway is 2.9 mrem/yr. This is based on the dose from effluent gaseous release calculated at the EAB using GASPAR II and then adjusting the dose for the Chi/Q in the worst meteorological sector and the location of the construction workers at HAR 3.
- No reduction in dose is applied for either the distances between the HNP Protected Area Boundary and HAR 2 or 3 construction sites, or any potential HNP direct radiation shielding by HAR 2 for HAR 3.

The direct radiation exposure was based on a 2,080-hour work year and an exposure rate of 14.8 µrem/hr yielding a total dose of 31 mrem/yr.

The annual collective dose to the construction workforce is estimated to be 120.3 person-rem (that is, the maximum individual dose multiplied by the number of people exposed). This estimate assumes 3,150 persons based on a construction worker dose of 38.2 mrem/yr.

The largest contributor to the total effective dose equivalent (TEDE) would be the external dose assumed from the active HNP operations (31 mrem/yr). Doses contributed by liquid effluents from HNP and HAR 2 provide an additional 0.8 mrem/yr. Doses contributed by gaseous effluents from HNP and HAR 2 provide an additional 6.4 mrem/yr. It is concluded that annual construction worker doses attributable to HNP operations for the proposed construction areas for HAR 2 and 3 are a fraction of those limits specified in 10 CFR Part 20 and 10 CFR Part 50 Appendix I.

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 worker worked 60 hours per week for 52 weeks the dose presented in Table 12.4-201 would increase by 50%. Even considering the most limiting case of 84 hours per week, 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.

12.4.1.9.5 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.

Add the following new subsection after DCD Subsection 12.4.3

12.4.4 REFERENCES

201. Progress Energy Carolinas, Inc. (PEC), “Shearon Harris Nuclear Power Plant Annual Radioactive Effluent Release Report: January 1, 2004 to December 31, 2004”.

202. Progress Energy Carolinas, Inc., “Shearon Harris Nuclear Power Plant Off-Site Dose Calculation Manual (ODCM),” Revision 17, Docket No. STN-50-400, November 30, 2004.

203. Progress Energy Carolinas, Inc. (PEC), "Shearon Harris Nuclear Power Plant Annual Radioactive Effluent Release Report: January 1, 2008 through December 31, 2008".

204. Nuclear Generation Group, "Area Thermoluminescent Dosimeter (TLD) Monitoring," DOS-NGGC-0010, Revision 7, 2006, Nuclear Generation Group Standard Procedure Volume 99 Book/Part 99, information obtained from the HNP TLD monitoring group via a request for information.

205. Progress Energy Carolinas, Inc. (PEC), "Shearon Harris Nuclear Power Plant Annual Radioactive Effluent Release Report: January 1, 2007 through December 31, 2007".

206. Progress Energy Carolinas, Inc. (PEC), "Shearon Harris Nuclear Power Plant Annual Radioactive Effluent Release Report: January 1, 2010 through December 31, 2010".

HAR SUP 12.4-1

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

Type of Radiation Dose	Public Dose Limits 10 CFR 20.1301	Estimated HAR Construction Worker Dose
Total effective dose equivalent (TEDE)	100 mrem/yr	Approximately 38.2 mrem/yr[a]
Maximum dose in any one hour	2 mrem	Less than 1 mrem

a) The largest contribution to the TEDE is from the external dose assumed from active HNP operations (31 mrem/yr). Doses contributed by liquid effluents from HNP and HAR 2 provide an additional 0.8 mrem/yr. Doses contributed by gaseous effluents from HNP and HAR 2 provide an additional 6.4 mrem/yr.

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

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

STD COL 12.3-1 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~~Section12.5~~ 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.

12AA.5.4.14 Groundwater Monitoring Program

STD COL 12.3-3

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 decommissioning.

Revise the REFERENCES section of NEI 07-03A, Reference 8, 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 of Access to 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

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 a 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.