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March 4, 2015
U7-C-NINA-NRC-150003

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
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South Texas Project
Units 3 and 4
Docket Nos. 52-012 and 52-013
Revised Response to Request for Additional Information

Reference: Letter, Scott Head to Document Control Desk, "Revised Response to Request for Additional Information," dated April 26, 2011, U7-C-STP-NRC-110070 (ML11119A076)

The referenced letter provided a revised response to NRC staff questions included in Request for Additional Information (RAI) letter number 407 related to the South Texas Project Units 3 and 4 Combined License Application (COLA) Part 2, Tier 2, Section 12.2. That response included a new supplement (Appendix 12B) to the STP 3 and 4 COLA which was subsequently incorporated into the COLA. This revised response to RAI question 12.02-20 replaces the previous response in its entirety.

Changes to the COLA where indicated by gray shading will be incorporated into the next routine revision of the COLA following NRC acceptance of the RAI response.

There are no commitments in this letter.

If you have any questions regarding these responses, please contact me at 979-316-3011 or Bill Mookhoek at 979-316-3014

D091
NRO

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 3/4/15

Scott Head
Manager, Regulatory Affairs
NINA STP Units 3 & 4

rhs

Attachment: RAI 12.02-20, Revision 2

cc: w/o attachment except*
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RAI 12.02-20

QUESTION:

The staff has reviewed the information provided in the revised response to RAI 12.02-14. On the basis of this review, the staff has the following questions, regarding this RAI.

1. The response includes STP 3&4 FSAR revisions to ABWR DCD Tables 12.2-5a and 12.2-5b as well as new Tables 12.2-31 (SFP Peak Gamma Radiation Source) and 12.2-32 (SFP Peak Neutron Source). These new tables provide the source term for the Spent Fuel Pool (SFP) based on 270% of the core load one day post shutdown. However, the response does not include a source term for radionuclides (corrosion and fission products) contained in the SFP water. This information is necessary to perform radiation dose calculations for workers standing on the fuel handling bridge and to completely satisfy Tables 12.2-5a and 12.2-5b of the ABWR DCD, which state that the SFP radiation source and geometry information is the applicant's responsibility. Also, Regulatory Guide 1.206, Section C.I.12.2.1 "Contained Sources", states that the applicant is to provide the models, parameters, and bases for all values used to calculate source magnitudes, for normal and accident conditions. Please provide this SFP water source term information and include it in the STP 3&4 FSAR.

2. A figure was included in the RAI response showing a vertical cut view of the SFP. Some of the dimensions on the figure appear to be inconsistent with the spent fuel storage shielding dimensions shown in Table 12.2-5c of the ABWR DCD and STP 3&4 FSAR. In addition, this figure does not include some of the dimensions and spatial relationships needed by the staff to perform a shielding calculation. Specifically, there appears to be an inconsistency between the depth of water above the fuel assemblies as shown in the SFP figure and the depth of water as indicated in Table 12.2-5c. In addition, the figure does not show the minimum distance between the top of a raised fuel assembly at maximum height and the SFP water surface, the height of the SFP fuel pool bridge above the water surface, and the location of refueling personnel during refueling operations on the fuel pool bridge. In order for the staff to be able to perform an accurate deep dose equivalent calculation for a person standing on the fuel pool bridge, please provide an updated figure of the SFP, addressing these items.

3. As stated in the response to 1. above, additional information is needed in order to perform a radiation dose calculation for workers standing on the fuel handling bridge. During refueling operation ANSI/ANS-57.1-1992 states that fuel handling equipment shall be designed so that the operator will not be exposed to >2.5 mrem/hr (25 μSv/hr) from an irradiated fuel unit, control component, or both, elevated to the up-position interlock with the pool at normal operating water level. Verify that the maximum dose rate to an operator on the fuel handling bridge will not exceed the 2.5 mrem/hr criteria stated in this ANSI/ANS standard.

4. Regulatory Guide 1.206, Section C.I.12.3.2, states that the applicant should provide information regarding the shielding of each radiation source. Section C.I.12.1.3 of this Regulatory Guide also states that applicants should describe how operating procedures will be used to ensure doses are kept ALARA. The RAI response does not provide a description of any design features/access controls that will be used to limit the dose rate to individuals working in the Spent Fuel Pool area. Please provide a description of any design features/access controls (such as installing additional shielding to the fuel handling bridge/refueling machine to lower dose rates to the operator on the

fuel handling bridge and minimizing the personnel stay times on the refueling bridge) that will be used, to ensure that the dose to refueling personnel are maintained ALARA during refueling operations.

5. ABWR DCD Chapter 12 figures provide radiation zone maps for areas adjacent to the spent fuel storage room. Please confirm that the dose rates in areas adjacent to the SFP room will be no higher than what is indicated in the ABWR DCD Chapter 12 figures and include any changes in estimated dose rates in the Chapter 12 FSAR radiation zone maps.

REVISED RESPONSE:

This response replaces in its entirety NINA's response to RAI 12.02-20, dated April 26, 2011, in letter U7-C-STP-NRC-110070. Changes to the referenced response are indicated by revision bars.

This response provides information regarding radiation dose rates from the Spent Fuel Pool (SFP). The response provides the necessary dimensions and calculated results for the dose rate to workers on the refueling machine trolley platform from the highest activity irradiated fuel bundle elevated to the maximum up-position, with water coverage of 8.5 ft from the top of the fuel assembly active fuel region. Additionally, tables are provided that list the peak gamma and neutron sources and radionuclides in the SFP. These tables are contained in FSAR Appendix 12B shown below.

1. Appendix 12B shows a sketch of the SFP model used to calculate doses in adjoining areas and at the surface of the pool (Figure 12B-1). The calculations were based on a preliminary rack configuration with a capacity of 2365 bundles, which is conservative for a dose calculation compared to the minimum 270% (2354) bundles required in the DCD. Subsequently, the final rack design has a capacity of 2380 assemblies. The additional 15 assemblies (0.63%) above the 2365 assemblies assumed in the calculations is not significant considering the 10% margin added to the source term and considering that these assemblies will be at least 10 years old. As discussed in the appendix, the radioisotope inventory and gamma and neutron sources, as a function of time post-irradiation, were calculated with the NRC-endorsed ORIGEN-S Code. The supporting proprietary calculations are referenced in this revised response as References 1 and 2.

   The Fuel Pool Cooling and Cleanup (FPC) system described in FSAR Subsections 9.1.3 and 12.3.1.4.3 maintains the SFP water at a very low radioactive nuclide level. In support of this statement, data from an operating ABWR plant has been used. This information is also provided in Appendix 12B in Section 12B.5 and Table 12B-7.

   FSAR Tables 12.2-5a and 12.2-5b will be revised as shown after the response to question 3 below to reflect the representative SFP geometry information. Gray highlighting shows the changes.

2. ABWR DCD Table 12.2-5c shows spent fuel storage with 7.4 m of water above the active fuel region of the fuel elements stored in the racks at the bottom of the pool in the fully seated storage position. BWR fuel with 62 fuel rods and an overall length of 176 inches has been supplied with active fuel regions from 144 inches to 150 inches in length. DCD Table 1.3-1 lists the active fuel region for the ABWR as 146 inches. This analysis assumes an active fuel

region of 3.8 meters (about 150 inches) which is characteristic of the high enrichment, high burnup fuel assumed in the analysis. The spent fuel racks are designed for 5.0 weight percent (wt%) maximum enrichment with integrated gadolinium in the fuel and 2.95 wt% maximum enrichment without gadolinium to maintain reactivity less than 0.95 for current and future fuel designs. DCD Table 1.3-1 lists an initial average U-235 enrichment of 2.22%, however, for conservatism; the calculation assumes an average enrichment of 4.2%, which, again, is characteristic of high burnup cores. The calculations assume a 24-month fuel cycle (higher burnup) than the 18-month cycle described in the DCD. In other words, in an 18-month cycle, the discharge batches would be smaller and more frequent, each batch would see less exposure, and the earlier batches would decay longer. Therefore, the 24-month high burnup fuel assumed in the analysis is conservative.

Figure 12B-1 in the previous RAI response showed the preliminary spent fuel pool rack layout. The dimensional data on that figure were not used in any calculations. Figure 12B-1 has been replaced with a sketch of the analytical model used in the calculations showing the critical dimensions.

As requested, the following information is provided. The refueling machine design discussed in Reference 4 shows the trolley platform approximately 8.8 ft (2.7 m) above the refueling floor. The refueling machine will have both local and remote controls. An improvement over some other machines allows the operator to locally control motions of the bridge, trolley, and all hoists and grapples from the trolley platform as opposed to operation of auxiliary hoists from the lower bridge. The remote control system consoles will be located in a control room on the refueling floor, providing the operator with all required functions for automatic control of fuel handling. As discussed in FSAR Subsection 9.1.4.1, the fuel grapple design will maintain adequate water shielding of at least 8.5 ft (2.6 m) over the top of the active fuel.

3. Section 12.2 of the ABWR DCD includes two items marked "applicant," in Tables 12.2-5a and 5b, for spent fuel source term and source term geometry. The NRC staff requested that these two "applicant" items (which are effectively another COL Information Item) be addressed in RAI 12.02-20. NINA's response to this RAI describes how NINA developed the source term and defined the geometry in response to these items. The following points are pertinent to these items:

- The "applicant" items required specific information be provided that was not directly available in the DCD, hence additional detailed analysis of the spent fuel was required. This was done as a stand alone assessment to address the "applicant" items and is being included in the COLA as supplemental information.

- The response to the "applicant" items is self contained and stands on its own. It is based on relevant information from the DCD (e.g. core thermal power, maximum burn up, # of spent fuel assemblies in the SFP, etc) and conservative assumptions (e.g. 2 year fuel cycle, 10% burn up) to develop an accurate but conservative basis for a spent fuel pool source term.

- In order to assess the source term of the fuel in the full spent fuel pool racks, it is necessary to determine the source for most of the spent fuel at long cooling times (up to 10 years) to generate enough spent fuel to fill the racks. That level of information is not available in DCD

Table 12.2-3b, which only provides overall core source terms with decay times up to one month. Thus a detailed analysis was required to develop that information.

- As part of this RAI it is also requested that an analysis for operator doses based on spent fuel movement operations be provided by STPNOC. DCD Table 12.2-3b only provides overall core source terms. In order to determine the maximum source term for any spent fuel assembly being moved by the fuel handling machine, it was necessary to model the core and extract the information for the most radioactive spent fuel assemblies. Therefore a stand alone response is required.

- NINA calculated the radiation dose rate from the highest activity irradiated fuel bundle as follows:

The STP 3 & 4 core is rated at 3926 MWt. It consists of 872 fuel bundles, 320 of which are installed as new assemblies each refueling. 3926MWt / 872 bundles equals a core average of 4.502 MWt per bundle. This average, however, consists of once, twice, and thrice burned bundles.

The highest radiation source is from a fuel bundle in the once-burned batch (also called first batch in Reference 1 of the References for this RAI response; see page 21), as is shown on Appendix Table 12B-6.

The STP 3&4 core design as discussed in Reference 1 (see page 21) indicates that the average burnup of the once-burned batch is approximately 23 GWd/MTU, or 5.703 MWt/bundle. This value is 1.27 times the average per bundle across the entire core.

To conform with the maximum relative assembly power peaking factor of 1.40 for the ABWR as reported in DCD Table 1.3-1, 10% was added to the calculated dose from the once-burned batch of 1.27 times the core average. The net effect is equivalent to using a bundle source at 1.4 times the core average. However, it is noted that the maximum relative assembly power peaking factor of 1.40 corresponds to an 18-month cycle core design. It is expected that a 24-month cycle core would result in a higher peaking factor than for the 18-month cycle core identified in the DCD. Westinghouse experience in the design of the ABWR core indicates a maximum radial peaking factor of 1.65 for a 24-month cycle core.

To ensure an absolute maximum activity bundle is reported, an additional 20% factor was added to the 10% factor, resulting in a bundle source of over 1.65 times the core average. The dose rate at the operating trolley platform from this highest activity fuel bundle, elevated to the maximum up-position in the SFP with water coverage of 8.5 ft (2.6 m) from the top of the fuel assembly active fuel region is approximately 1.2 mrem/hr. This dose rate from the maximum activity bundle with 30% factor added is clearly bounding for any assembly from the 18-month cycle core with a 1.40 peaking factor that will be moved by the refueling operator, and is appreciably below the criteria of 2.5 mrem/hr (25 μSv/hr) stated in ANSI/ANS - 57.1 – 1992, subsection 6.3.4.1.5. The supporting proprietary calculation is discussed in Reference 3 (see page 21). This dose rate, to an operator on the trolley platform at 8.8 ft (2.7 m) above the refueling floor, would be considerably less if credit is given to reduction because of the additional 1 ft (0.3 m) of air space between the water surface and refueling floor, and attenuation through the refueling machine lower structure and platform.

12.4.2 Reactor Building Dose

STD DEP 9.1-1

STD DEP Admin

The following provides the basis by which the Reactor Building dose estimates for occupational exposure were made.

(2) *ABWR refueling is accomplished via an automated refueling* ~~*bridge*~~ machine. *All operations for refueling are accomplished* ~~*from an enclosed automation center off the refueling floor*~~ as described in Section 9.1.4.2.7.1. *Time for refueling is reduced from a typical 4,400 person-hours down to 2,000 person*-hours *and from an effective dose rate of 25* ~~*μGy/h*~~μSv/h *to less than 2* ~~*μGy/h*~~μSv/h. The dose rate from the highest activity irradiated fuel bundle elevated to the maximum up-position in the spent fuel pool (SFP) with water coverage of 8.5 ft (2.6 m) from the top of the fuel assembly active fuel was calculated. The dose rate at the refueling machine trolley platform 8.8 ft (2.7 m) above the refueling floor is approximately 1.2 mrem/hr (12 μSv/hr). This dose rate of 1.2 mrem/hr (12 μSv/hr) is conservative because of the additional 1 ft (0.3 m) of air space between the water surface and refueling floor, and the attenuation through the refueling machine lower structure and platform. Additionally, it is below the criteria of 2.5 mrem/hr from an irradiated fuel unit, control component, or both, as required by ANSI/ANS-57.1-1992. Note that this value does not include dose from radionuclides contained in Spent Fuel Pool water, which is expected to be no greater than 0.7 mrem/hr. at approximately 1.2 meters above the Refueling Floor, based on data from currently operating plants utilizing the ABWR Spent Fuel Pool design. This value takes no credit for attenuation through Refueling Machine construction material or the additional distance to the Refueling Machine trolley platform, which adds additional conservatism.

Table 12.2-5a Radiation Sources—Radiation Sources

Source Table	For	Drawing	Location	Approximate Geometry
~~*Applicant*~~ **Appendix 12B**	Spent Fuel Storage	12.3-6 **12.3-10**	*(R4,RF)* **RF**	*See* ~~*Drawings*~~ ~~***~~ **Appendix 12B**

~~*** Applicant to develop spent fuel storage facilties design drawings showing geometry of facilities.~~

Table 12.2-5b Radiation Sources—Source Geometry

Component	Assumed Shielding Source Geometry
Spent Fuel Storage	~~*Applicant* ***~~ **Homogeneous source over the assumed active fuel volume of racks in the pool**

~~*** Applicant to develop spent fuel storage facilties design drawings showing the shielding source geometry.~~

12B Spent Fuel Pool Geometry and Dose Rate Calculation

This Appendix provides ~~site specific~~ information regarding the ~~Spent Fuel Pool (SFP) geometry and~~ methodology used to generate Spent Fuel Pool (SFP) radiation sources and dose rate assessments. ~~This appendix supplements the SFP radiation asource information in Section 12.2, specifically Tables Tables 12.2-5a and 12.2-5b.~~ See FSAR Section 12.2, Tables 12.2-5a, 12.2-5b, and 12.2-5c; and FSAR Section 12.3, Figures 12.3-5 through 12.3-10. ~~The SFP storage rack has the capacity for 2365 BWR fuel assemblies. The size and configuration of the storage racks are given in Figure 12B-1, which shows an irregular cross-section areaextended tp 10.7 m x 9.0 m. The fuel storage rack is situated at the lowest level of the SFP, where the total height of water is 11.6 m.~~

12B.1 Core Source Determination

The STP 3 & 4 core rated power is 3926 MWt. The core load consists of 872 fuel assemblies, with an assumed average enrichment of 4.2%. The reload batch size is 320 fuel assemblies. The equilibrium core is assumed to operate on a two-year fuel cycle, with approximately 704 effective-full-power-days (EFPD) of operation for each cycle. The core is composed of three batches of fuel, with varying residence time and burnup at the end of cycle (EOC). The latest batch of fresh fuel assemblies will be placed at high power density locations during operation and will average 23 GWD/T at EOC, 1.27 times the core average (18 GWD/t). The second oldest batch will have resided in the core for one cycle prior to the current cycle and will have a cycle exposure of 20 GWD/t (1.14 times average) and an average accumulated exposure of 43 GWD/T at EOC. The oldest batch will have a cycle exposure of 8 GWD/t (0.43 times average) and an average accumulated exposure of ~~resided in the core for two cycles prior to the current cycle and on average accumulated~~ 51 GWD/T at EOC. Radiation sources of the equilibrium core, as well as the discharge batch, are calculated using the code ORIGEN-S of the SCALE system (Reference 12B.6-1).

In addition to fuel elements, the mass of supporting structural materials is also included in the ORIGEN-S calculations. Therefore, the calculated radiation source includes contributions from activation and corrosion products as well as fission products and actinides. In addition, a safety margin of 10% is added to the calculated radiation source to bound the uncertainties in structural material specification. The resulting equilibrium core gamma source is shown in Table 12B-1.

12B.2 SFP Fuel Source Determination

The SFP radiation source is calculated using the maximum capacity of the storage rack, five batches of spent fuel discharged at two-year intervals plus a full core discharge. ~~Under normal operating conditions, the maximum number of fuel assemblies accommodated by the storage rack is 1493, leaving enough room for a full core offloading the event of unexpected operating conditions.~~ The maximum SFP radiation source when the storage rack is filled to its capacity includes the full equilibrium core radiation source conservatively assumed to be at one-day post shutdown plus the sources of existing fuel assemblies in the SFP. The existing fuel assemblies in the SFP are assumed to be the discharged batches from an equilibrium core at decay times of two-years, four-years, six-years, eight-years, and ten-years. Calculation of these sources has been described in 12B.1 above. Similar to the core radiation source, an extra 10% safety margin is added to the calculated results for conservatism. The resulting SFP peak gamma source and peak neutron source are given in Tables 12B-2 and 12B-3, respectively. The major radionuclides that provide the peak SFP source are listed in Table 12B-4.

12B.3 SFP Dose Rate Assessment

The SFP radiation source is used for the shielding design and analysis to ensure radiation levels at neighboring areas surrounding the SFP meet the design criteria. A simplified model of a rectangular parallelepiped of (8.1 m x 8.1 m x 11.2 m) is assumed to represent the SFP. The SFP peak radiation source is assumed to be homogenized over the bottom 3.8 m of the SFP to represent the active fuel length of a typical BWR fuel assembly. 3.8 meters was assumed to represent the height of the active fuel region, since this height results in the minimum required shielding of water, 11.2 m – 3.8 m = 7.4 m (see Table 12.2-5c). The~~re are 2 m thick~~ concrete walls and floor surrounding the SFP ~~on all sides. The floor of the SFP is~~ are 2 meters thick.~~ as well.~~(see Table 12.2-5c). The water shielding and distance between the spent fuel racks and the surrounding walls in the pool are not considered. (See Figure 12B-1). The dose rate calculations are carried out with the point-kernel shielding code QAD-CGGP-A (Reference 12B.6-2). The calculated dose rate results indicate all the areas surrounding the SFP meet the reactor building (R/B) radiation zone limits. Figure 12B-2 shows the dose rate profile for a set of detectors along the centerline of the SFP perpendicular to the water surface. The ~~D~~dose rate at the water surface is approximately 0.001 μRem/hr (1E-5 μSv/hr). The dose rates assessed are due to fuel assemblies in the SFP alone. Contributions from contaminants in the SFP water are not included.

12B.4 Single Bundle Refueling Operator Dose Rate Assessment

The maximum calculated radiation dose rate to the refueling operator from a single raised fuel assembly is calculated as follows:

As discussed in 12B.1 above, the core is rated at 3926 MWt. It consists of 872 fuel bundles, 320 of which are installed as new assemblies each refueling. 3926MWt / 872 bundles equals a core average of 4.502 MWt per bundle. This average, however, consists of once, twice, and thrice burned bundles.

Similar to the SFP dose rate assessment, QAD-CGGP-A is used for the single bundle dose rate calculations. Three separate calculations are performed to address the contributions from (1) the high enrichment lower active fuel region, (2) the low enrichment (natural uranium) upper active fuel region (top node) where the power density is lower than the core average, and (3) the upper non-active fuel assembly plenum (referred to as the handle) ~~the active fuel region, the top fuel node where the power density is lower than the core average, and the assembly handle~~ where it is assumed that majority of the activation products accumulate. The radiation source input for each of these components and the fuel assembly dimensions in the calculation model are presented in Table 12B-5 (the values in this table do not include an added safety margin).

DCD 9.1.4.1 requires that a fully retracted fuel grapple must maintain 2591 mm (8.5 ft) of water shielding over fuel. Therefore, the model assumes that the top node is just below the 8.5 feet of shielding water, and the handle element is just above that level. The movement of a raised fuel assembly from a once burned batch results in the highest dose rate to a worker on the refueling platform during movement of the assembly from the reactor core to the spent fuel pool.

The core design indicates that the average burnup of the once-burned batch is approximately 23 GWd/MTU, or 5.703 MWt/bundle. This value is 1.27 times the average per bundle across the entire core, as stated above. This factor would have to be increased by 10% to reflect a peak bundle of 1.4 times the core average (see DCD Table 1.3-1). However, the DCD radial peaking factor of 1.4 is for an 18-month cycle. To account for the 24-month cycle assumed in this assessment, a radial peaking factor for a 24-month core design of 1.65 is typical. Consequently, a factor of 30% was applied to the 1.27 times core average value to reflect the 1.65 peaking factor. ~~To obtain the highest activity bundle for determining the refueling source term, an additional 30% was added to the calculated dose rate from the once burned batch of 1.27 times the core average. The net effect is equivalent to using a bundle source of 1.65 times the core average for this highest activity fuel assembly.~~ The dose rate is bounding for any assembly that will be moved by the refueling operator. (The calculated gamma source of a single bundle with one-cycle residency is compared with that of three-cycle residency in Table 12B-6, which further confirms that peak source has been chosen for the refueling dose rate assessment. The values in this table do not include an added safety margin.)

It is assumed that the fuel assembly is lifted to a height of 8.5 ft (2.6 m) below the pool water surface with the operator on the refueling machine trolley platform at a minimum of 8.8 ft (2.7 m) above the water surface. The resulting peak dose rate at 8.8 ft (2.7 m) above the water surface is approximately 1.2 mRem/hr, located at a radial distance from the fuel assembly of approximately 140 cm (4.6 ft) as shown in Figure 12B-3. Even when the maximum fuel pool water source (see Section 12B.5 below) is added, the dose rate remains below 2.5 mRem/hr. For an operator standing on the trolley platform, the dose rate will be less than that shown in Figure 12B-3; therefore the design criteria of ANSI/ANS-57.1-1992 has been satisfied.

12B.5 Spent Fuel Pool Radionuclides and Dose

The Fuel Pool Cooling and Cleanup (FPC) system described in FSAR Subsections 9.1.3 and 12.3.1.4.3 maintains the SFP water at a low radioactive nuclide level. In support of this statement, representative data from an operating ABWR plant is presented. Measurements at 1.2 m above the refueling floor indicate a maximum of 0.007 mSv/hr (0.7 mrem/hr) during plant outages with 1) fuel assemblies fully seated in the storage racks, and 2) maximum levels of measured radionuclides in the SFP water. This maximum dose rate is measured at 1.2 m (3.9 ft) above the refueling floor. During routine operations, the dose rate is expected to be less for normal operation of the FPC system. Also, the dose rate is considerably less at the operating trolley platform due to the increased distance from 1.2 m (3.9 ft) to 2.7 m (8.8 ft) and attenuation through the refueling machine lower structure and platform. A listing of SFP water radionuclides for a representative ABWR is summarized in Table 12B-7.

12B.6 References

12B.6-1 NUREG/CR-0200, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," Rev 7, May 2004.

12B.6-2 CCC-645, "QAD-CGGP-A: Point Kernel Code System for Neutron and Gamma-Ray Shielding Calculations Using the GP Buildup Factor," Oak Ridge National Laboratory, December 1995.

12B.6-3 CN-REA-10-53, "STP Units 3 & 4 ABWR Spent Fuel Pool Radiation Source", Revision 1, Westinghouse Electric Company LLC, December 15, 2010.

12B.6-4 CN-REA-10-64, "STP Units 3 & 4 ABWR Spent Fuel Dose rates", Revision 0, Westinghouse Electric Company LLC, October 15, 2010

12B.6-5 CN-REA-10-67, "Dose rate Evaluation from a Single ABWR Fuel Bundle in STP Units 3 & 4 Spent Fuel Pool", Revision 0, Westinghouse Electric Company LLC, October 10, 2010

12B.6-6 LTR-ABWR-LIC-11-001, "Scaling of Dose rate for a Single Fuel Bundle Dose to Worker on Refueling Platform – STP 3&4", March 31, 2011

Table 12B-1: Gamma Source* of an Equilibrium Core vs. Time Post-Shutdown

Energy Range (MeV)	Gamma Source (MeV/s-MW)	
	1-Day	30-Day
0.02 - 0.035	1.4E+14	2.7E+13
0.035 - 0.05	1.3E+14	3.2E+13
0.05 - 0.075	1.1E+14	2.6E+13
0.075 - 0.125	1.6E+15	5.5E+13
0.125 - 0.175	4.5E+14	1.2E+14
0.175 - 0.25	1.0E+15	3.6E+13
0.25 - 0.40	1.9E+15	1.4E+14
0.40 - 0.90	1.1E+16	3.9E+15
0.90 - 1.35	1.5E+15	2.1E+14
1.35 - 1.80	3.4E+15	6.9E+14
1.80 - 2.20	2.5E+14	5.3E+13
2.20 - 2.60	2.5E+14	5.1E+13
2.60 - 3.00	4.9E+12	1.1E+12
3.00 - 3.50	1.9E+12	4.5E+11
3.50 - 4.00	1.3E+10	3.3E+07
4.00 - 4.50	2.1E+09	2.5E+06
4.50 - 5.00	1.8E+10	1.6E+06
5.00 - 10.00	1.3E+08	2.8E+06
Total	2.2E+16	5.4E+15

* Based on rated power 3926 MWt and approximately 704 effective-full-power-days (EFPD) of operation each cycle. 10% safety margin added to ORIGEN-S results.

Table 12B-2: SFP Peak Gamma Source (MeV/s) vs Time Post-Shutdown

Energy Group	Energy Range (MeV)	1-Day	1-Year	2-Year
1	0.02 - 0.035	5.5E+17	3.9E+16	2.5E+16
2	0.035 - 0.05	5.1E+17	4.1E+16	2.3E+16
3	0.05 - 0.075	4.5E+17	4.1E+16	2.4E+16
4	0.075 - 0.125	6.2E+18	8.5E+16	5.1E+16
5	0.125 - 0.175	1.8E+18	8.3E+16	4.1E+16
6	0.175 - 0.25	3.9E+18	5.6E+16	3.2E+16
7	0.25 - 0.40	7.4E+18	1.2E+17	6.8E+16
8	0.40 - 0.90	4.5E+19	3.6E+18	2.5E+18
9	0.90 - 1.35	6.0E+18	2.1E+17	1.5E+17
10	1.35 - 1.80	1.3E+19	8.3E+16	5.1E+16
11	1.80 - 2.20	9.9E+17	4.7E+16	2.0E+16
12	2.20 - 2.60	9.7E+17	4.1E+15	2.0E+15
13	2.60 - 3.00	1.9E+16	7.3E+14	3.6E+14
14	3.00 - 3.50	7.6E+15	1.0E+14	5.1E+13
15	3.50 - 4.00	5.2E+13	1.1E+11	7.0E+10
16	4.00 - 4.50	8.4E+12	2.4E+10	2.3E+10
17	4.50 - 5.00	6.9E+13	1.6E+10	1.5E+10
18	5.00 - 10.00	5.4E+11	2.7E+10	2.5E+10
Total		8.8E+19	4.4E+18	3.0E+18

Note: The data represent one full core offload plus existing pool batches. 10% safety margin is added.

Table 12B-3: SFP Peak Neutron Source

Energy Group	Neutron Energy (KeV)	(n/s) 1-Day Post-Shutdown
1	1.0E-08 – 1.0E-05	1.0E-02
2	1.0E-05 – 3.0E-05	8.0E-03
3	3.0E-05 – 5.0E-05	1.4E-00
4	5.0E-05 – 1.0E-04	1.2E-00
5	1.0E-04 – 2.25E-04	9.5E+00
6	2.25E-04 – 3.25E-04	8.7E+00
7	3.25E-04 – 4.0E-04	7.9E+00
8	4.0E-04 – 8.0E-04	5.1E+01
9	8.0E-04 – 1.0E-03	2.7E+01
10	1.0E-03 – 1.13E-03	2.6E+01
11	1.13E-03 – 1.3E-03	3.1E+01
12	1.3E-03 – 1.77E-03	9.7E+01
13	1.77E-03 – 3.05E-03	3.3E+02
14	3.05E-03 – 0.01	2.9E+03
15	0.01 – 0.03	1.5E+04
16	0.03 – 0.1	9.3E+04
17	0.1 – 0.55	1.3E+06
18	0.55 – 3.0	1.7E+07
19	3.0 – 17.0	2.3E+08
20	17.0 – 100.0	3.2E+09
21	100 – 400	2.1E+10
22	400 – 900	4.7E+10
23	900 – 1400	4.7E+10
24	1400 – 1850	3.8E+10
25	1850 – 3000	7.2E+10
26	3000 – 6430	6.5E+10
27	6430 – 2.0E+04	5.9E+09

Note: The data represent one full core offload plus existing pool batches. 10% safety margin is added.

Table 12B-4: Peak Source Radioisotopes in the Spent Fuel Assemblies

Isotopes	Curies	Isotopes	Curies
I-131	1.07E+08	Sr-92	3.25E+05
I-132	1.36E+08	Y-91	1.46E+08
I-133	1.09E+08	Y-91M	1.56E+07
I-134	6.20E+00	Y-92	4.81E+06
I-135	1.79E+07	Y-93	3.35E+07
Total I	**3.71E+08**	Zr-93	3.78E+02
		Zr-95	2.02E+08
Na-24	3.19E+03	Nb-95M	2.24E+06
P-32	3.16E+04	Nb-95	1.99E+08
Cr-51	1.76E+07	Mo-99	1.67E+08
Mn-54	7.62E+05	Tc-99M	1.60E+08
Mn-56	6.13E+04	Tc-99	2.61E+03
Fe-55	6.38E+06	Ru-103	1.75E+08
Fe-59	3.88E+05	Rh-103M	1.75E+08
Co-58	9.22E+05	Ru-106	6.63E+07
Co-60	1.54E+05	Rh-106	6.63E+07
Ni-63	1.71E+05	Ag-110M	3.50E+05
Cu-64	1.59E+04	Ag-110	4.77E+03
Zn-65	3.70E+01	Te-129M	5.68E+06
Sr-89	1.11E+08	Te-129	5.21E+06
Sr-90	1.51E+07	Te-131M	1.44E+07
Y-90	1.55E+07	Te-131	2.89E+06
Sr-91	2.46E+07	Te-132	1.32E+08
Total	**1.93E+08**	Ba-137M	1.89E+07
		Ba-140	1.93E+08
Kr-83m	6.41E+04	La-140	2.06E+08
Kr-85m	7.52E+05	Ce-141	1.90E+08
Kr-85	1.89E+06	Ce-144	1.61E+08
Kr-87	1.27E+02	Pr-144M	2.25E+06
Kr-88	2.30E+05	Pr-144	1.61E+08
Total Kr	**2.94E+06**	W-187	1.91E+05
		Np-239	1.65E+09
Xe-131m	1.25E+06	Pu-239	5.03E+04
Xe-133m	6.63E+06	**Total**	**4.16E+09**
Xe-133	2.28E+08		
Xe-135m	2.92E+06	Cs-134	2.65E+07
Xe-135	5.84E+07	Cs-135	8.82E+01
Total Xe	**2.98E+08**	Cs-136	5.21E+06
Noble Gas Totals	**3.00E+08**	Cs-137	2.01E+07
		Total Cs	**5.18E+07**

Note: The data represent one full core offload plus existing pool batches. 10% safety margin is added.

Table 12B-5 Single Bundle Source for Dose Assessment (Photons/s)

Energy Group	Energy Range (MeV)	Enriched Fuel Region	Top Node	Handle
1	0.02 - 0.035	2.5E+16	1.9E+14	6.0E+13
2	0.035 - 0.05	1.5E+16	1.1E+14	7.4E+12
3	0.05 - 0.075	8.4E+15	7.0E+13	8.7E+12
4	0.075 - 0.125	7.0E+16	1.1E+15	7.4E+12
5	0.125 - 0.175	1.6E+16	1.1E+14	2.2E+13
6	0.175 - 0.25	2.2E+16	2.7E+14	3.4E+12
7	0.25 - 0.40	2.8E+16	3.2E+14	3.5E+13
8	0.40 - 0.90	8.7E+16	6.4E+14	4.8E+14
9	0.90 - 1.35	6.0E+15	4.2E+13	1.6E+13
10	1.35 - 1.80	1.1E+16	8.0E+13	1.6E+12
11	1.80 - 2.20	4.9E+14	3.4E+12	4.2E+11
12	2.20 - 2.60	5.4E+14	3.7E+12	1.2E+10
13	2.60 - 3.00	8.9E+12	6.6E+10	5.5E+10
14	3.00 - 3.50	3.2E+12	2.2E+10	1.6E+09
15	3.50 - 4.00	2.1E+10	1.3E+08	2.5E+07
16	4.00 - 4.50	3.1E+09	1.7E+07	1.7E+04
17	4.50 - 5.00	2.3E+10	1.3E+08	1.3E+05
18	5.00 - 10.00	1.1E+08	6.0E+05	6.0E+02
Total		2.9E+17	2.9E+15	6.4E+14
Source Height (cm)*		366	15	15

* Bundle cross-section is 15.24 cm x 15.24 cm (6 in x 6 in).

Table 12B-6 Single Bundle Gamma Source (MeV/s) Comparison

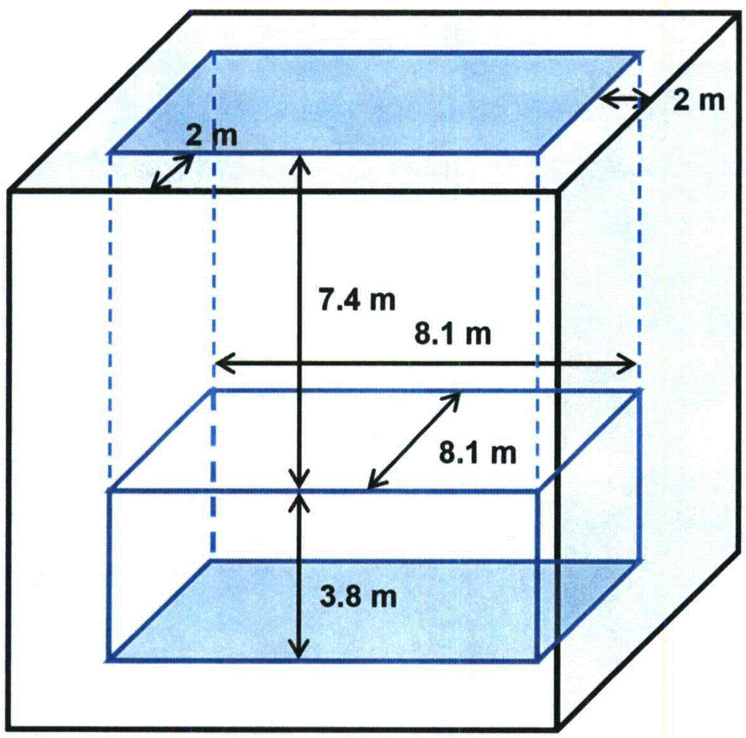
Decay Time	Once Burned	Thrice Burned
1 Day	1.1E+17	5.8E+16
30 Days	2.6E+16	1.5E+16
1 Year	2.1E+15	3.6E+15
2 Years	1.0E+15	2.4E+15
3 Years	7.0E+14	1.8E+15

Table 12B-7 SFP Water Radionuclides

Nuclide	Bq/cm^3	Nuclide	Bq/cm^3
Cr-51	9.4E-02	Nb-95	8.1E-03
Mn-54	9.7E-02	Cs-134	1.4E-02
Mn-56	3.6E-02	Cs-137	7.7E-03
Co-58	5.1E-01	Sb-124	5.1E-01
Co-60	1.3E+00	Sb-125	3.4E-02
Cu-64	1.0E-01		

Figure 12B-1 Spent Fuel Pool Analytical Model

8.1 m x 8.1 m x 3.8 m Active Fuel Volume
8.1 m x 8.1 m x 7.4 m Water Shielding Volume
2.0 m Thick Walls and Floor



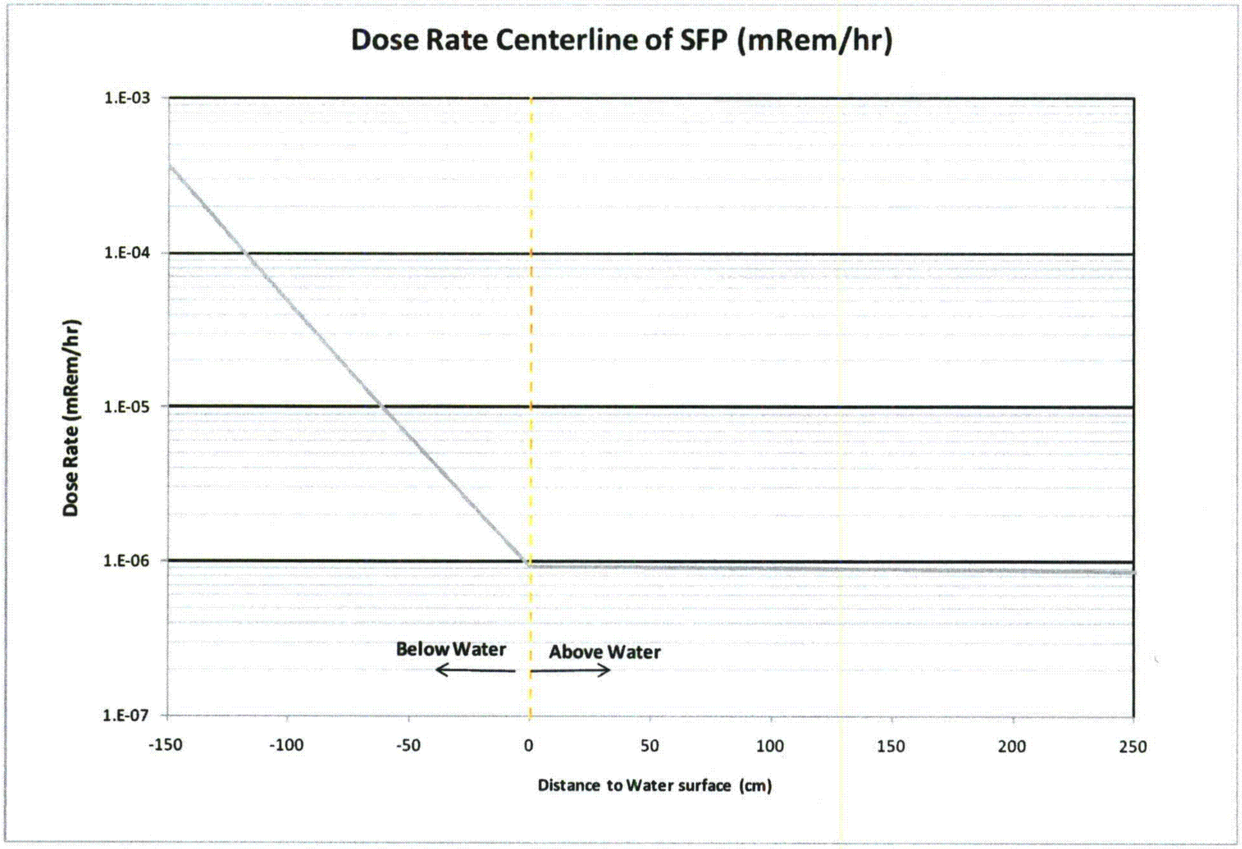
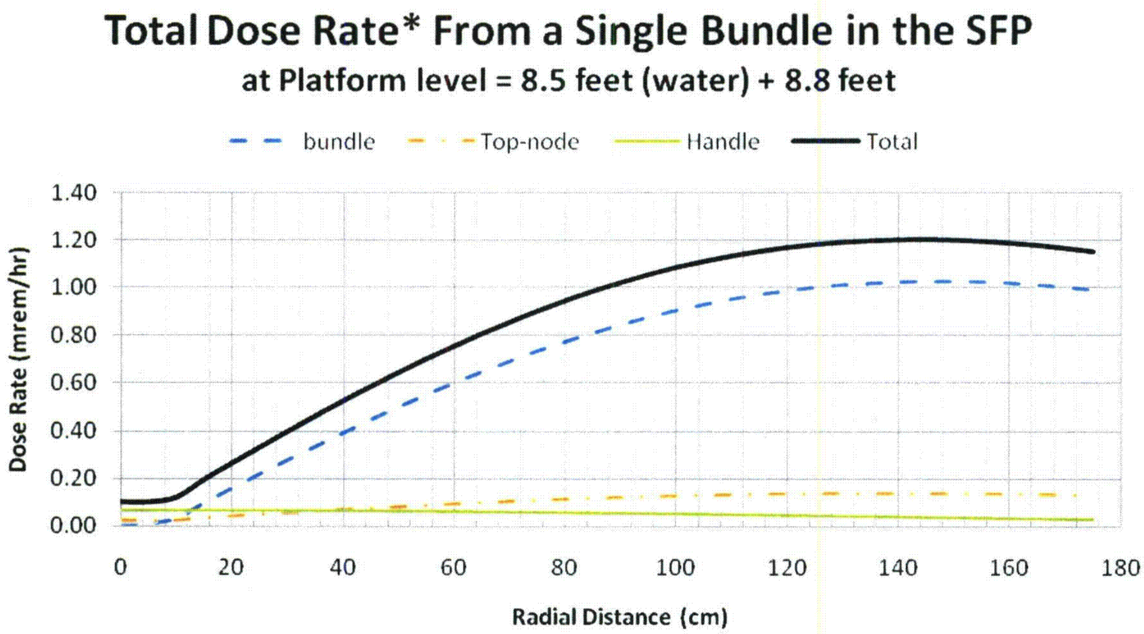


Figure 12B-2 SFP Centerline Dose Rate



~~* 30% safety margin is added~~

Figure 12B-3 Maximum Single Bundle Dose Rate at Refueling Platform

4. The STP 3&4 Operational Radiation Protection Program is addressed in the FSAR. Specifically, in Section 12.5S, Nuclear Energy Institute Report NEI 07-03A, "Generic FSAR Template Guidance for Radiation Protection Program Description" and NEI 07-08A, "Generic FSAR Template Guidance for Ensuring that Occupational Radiation Exposures are as Low as is Reasonably Achievable (ALARA)," are incorporated by reference. The guidance contained in these two documents will be used to develop and implement the Radiation Protection and ALARA programs for STP 3&4, including refueling and spent fuel handling activities.

   NEI 07-03A provides the following sections that include guidance for procedures addressing shielding and dose to workers:

   - Section 12.5.4.1 "Radiological Surveillance"
   - Section 12.5.4.2 "Methods to Maintain Exposures ALARA"
   - Section 12.5.4.4 "Access Control"
   - Section 12.5.4.5 "Radiation Work Permits"
   - Section 12.5.4.6 "Personnel Monitoring"
   - Section 12.5.4.7 "Dose Control"
   - Section 12.5.4.9 "Respiratory Protection"
   - Section 12.5.4.11 "Radiation Protection Training"

   NEI 07-08A provides the following guidance for ensuring that occupational radiation exposures are ALARA in Section 12.1.3:

   - Radiation protection training, the radiation protection plan, the RWP system, and procedure reviews all help to ensure that radiation exposure of personnel is maintained ALARA. The following examples illustrate the incorporation of ALARA work practices.
   - Personnel required to be monitored for radiation exposure in accordance with 10 CFR 20.1502 are assigned TLDs to establish exposure history.
   - Workers are provided with direct-reading dosimeters on jobs, so that the worker can determine accumulated exposure at any time during a job.
   - Dose rate meters are used as needed to identify elevated dose rates.
   - Pre-job briefs are used to review radiological surveys and to plan work before personnel enter a radiation area. Written procedures provide guidelines regarding the amount of detail to be included in the pre-job briefings.
   - For work involving high radiation areas, high collective doses, high levels of removable contamination relative to site posting criteria for contamination areas, or known or suspected airborne radioactivity areas.

- Work is preplanned to minimize personnel exposure as defined in ALARA program procedures.

  - Radiation protection personnel provide coverage as required by radiation protection procedures.
  - On complex jobs in high radiation areas, dry-run training may be utilized. In some cases, mockups are used to familiarize workers with the operations that they are to perform. These techniques are beneficial to improving worker efficiency and minimizing the amount of time spent in the radiation field.
  - As practical, work area entry and exit points are established in areas with low radiation levels. This is done to minimize dose accumulated while changing.
  - Protective clothing and respiratory equipment: Control points are also established to minimize the spread of removable contamination from the job site.
  - Individuals working in radiologically controlled areas are trained to be aware of the varying intensities of radiation fields within the general vicinity of their job locations, and are instructed to remain in the areas of lower radiation levels as much as possible, consistent with performing their assigned tasks.
  - For high radiation area jobs, maps, postings, and/or detailed instructions are provided to clearly delineate the source of radiation or to alert personnel concerning the location of elevated dose rates. Provided with this information, workers will be cognizant of their immediate radiological environment and will minimize their stay times in areas of elevated dose rates, thus maintaining exposures ALARA.
  - Protective clothing and respiratory equipment prescribed by radiation protection personnel are commensurate with the radiological hazards involved. These requirements cannot be modified without the permission of radiation protection personnel. Consideration is given to the discomfort of workers to minimize the effect of protective efforts on efficiency and the time spent in a radiation area.
  - Special tools or jigs are used on jobs when their use permits the job to be performed more efficiently or prevents errors, thus reducing the time spent in a radiation area.
  - Where applicable, special tools are used to increase the distance from the source to the worker, thereby reducing the exposure received.
  - Consideration is given to the use of remote monitoring of personnel with various combinations of audio, visual, and dose information to reduce exposure of personnel. Direct communications (e.g., radios) may be used to further enhance radiation protection.
  - Permanent shielding is used, where practicable, to reduce radiation exposure at the work site and in designated "waiting areas" for personnel during periods when they are not actively involved in the work.
  - On some jobs, temporary shielding such as lead sheets draped or strapped over a pipe or concrete blocks stacked around a piece of equipment is used. Temporary shielding is used only if the estimated total exposure, which includes exposure received during installation and removal, is reduced. Experience with such operations is used in developing guidelines in this area.

These program components are discussed primarily in FSAR Sections 12.3, 12.5, 12.5S, 13.1, and 13.2, and elsewhere throughout the FSAR.

Specifically, for fuel handling in the SFP area, this RAI requests information regarding shielding, a description of how operating procedures will be used to ensure doses are kept ALARA, and a description of design features/access controls that will be used to limit the dose rate to individuals working in the SFP area.

The above discussion of the STP 3&4 Operational Radiation Protection Program fully addresses procedures to ensure doses are kept ALARA.

As discussed in the response to 3. above, the 8.5 ft (2.6 m) water cover from the top of the active maximum irradiated bundle is sufficient to ensure the dose rate to refueling operators is less than 2.5 mrem/hr. Note that administrative controls will require temporary shielding as warranted to maintain doses ALARA. However, installation of additional shielding to the refueling machine is not warranted. An improvement over some other machines is that the operator locally controls motion of the bridge trolley, and all hoists and grapples at the trolley platform as opposed to operation of auxiliary hoists from the lower height bridge. Access control, monitoring, and stay times for personnel in the refueling area will be procedurally controlled with radiation work permits (RWPs) required as also discussed above.

Another design provision in addition to the shielding, automated refueling handling machine, radiation monitoring and alarms, and controls is the design of the Fuel Pool Cooling and Cleanup (FPC) system described in FSAR Subsections 9.1.3 and 12.3.1.4.3. This system reduces radiation dose rates and further clarifies water for increased visibility, thus reducing worker stay time for fuel movement activities. FSAR Table 12.2-16 reflects the radionuclide removal capability of the FPC demineralizers.

No COLA change is required as a result of this response.

5. ABWR DCD Tier 2 Figure 12.3-6, Reactor Building Radiation Zone Map for Full Power & Shutdown Operation at Elevation 18100mm and Figure 12.3-10, Reactor Building Radiation Zone Map for Full Power and Shutdown Operation, Section A-A, provide the radiation zone maps for areas adjacent to the SFP. NINA has calculated the radiation dose rates in these areas by modeling the 270% full core loading in close proximity to the SFP walls for a conservative dose rate to the adjacent rooms. The dose rates were calculated using the point-kernel code, QAD, with a concrete wall thickness of 2 m consistent with DCD Table 12.2-5c, with results as follows. The calculated values were increased by 10% to provide an added safety margin. The supporting proprietary calculation is referenced below as Reference 2.

Room/Area	DCD Figure	Location	DCD Zone Dose Rate	Calculated Dose Rate
Stairwell Large Component Hatch	12.3-6	R5, RF	<5 mrem/hr (< 50 μSv/h)	< 0.71 mrem/hr
Miscellaneous Electrical Component Area	12.3-6	R3, RF	<5 mrem/hr (< 50 μSv/h)	< 0.36 mrem/hr
Below SFP (FCS Room, Penetration Room, Upper Drywell).	12.3-10	RE, RF, RG	<5 mrem/hr (< 50 μSv/h)	< 0.35 mrem/hr

The above results demonstrate that the dose rates in rooms/areas adjacent to the SFP are bounded by those given in the DCD.

No COLA change is required as a result of this response.

References for RAI 12.02-20 Response

The following references support information provided in this response and can be made available for NRC review upon request.

1. CN-REA-10-53, Revision 1, STP Units 3&4 ABWR Spent Fuel Pool Radiation Source, Wang S., December 15, 2010.

2. CN-REA-10-64, Revision 0, STP Units 3&4 ABWR Spent Fuel Pool Dose Rates, Wang S., October 15, 2010.

3. CN-REA-10-67, Revision 0, Dose Rate Evaluation from a Single ABWR Fuel Bundle in STP Units 3&4 Spent Fuel Pool, Song, P., October 10, 2010.

4. U7-RB-M-SPEC-RFM-0001. STP Units 3&4 Equipment Requirement Specification for Refueling Machine, Revision B, Filing Number RS-5129772, December 26, 2008.

5. LTR-ABWR-LIC-11-001, "Scaling of Dose Rate for a single Fuel Bundle Dose to Worker on Refueling Platform-STP 3 & 4" March 31, 2011