RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**7-7855**
SRP Section:	**12.02 – Radiation Sources**
Application Section:	**12.2, 12.3, and 9.1.4**
Date of RAI Issued:	**05/14/2015**

Question No. 12.02-1

10 CFR 52.47(a)(5) requires that the FSAR contain the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in 10 CFR 20.

10 CFR 50, Appendix A, Criterion 61, requires that the fuel storage and handling, radioactive waste, and other systems which may contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions, with suitable shielding for radiation protection, and with appropriate containment, confinement, and filtering systems.

SRP Section 9.1.4 indicates that acceptance for meeting the relevant aspects of GDC 61 is based in part on the guidelines of American National Standards Institute/American Nuclear Society (ANSI/ANS) 57.1-1992.

SRP Section 12.2 indicates that radiation sources should be determined and provided for all radiation sources that require (1) shielding, (2) special ventilation systems, (3) special storage locations and conditions, (4) traffic or access control, (5) special plans or procedures, or (6) monitoring equipment. The source descriptions should include all pertinent information required for (1) input to shielding codes used in the design process, (2) establishment of related facility design features, (3) development of plans and procedures, (4) assessment of occupational exposure and (5) determination of radiation dose to electrical equipment important to safety as described in 10 CFR 50.49.

SRP Section 12.2 also indicates that source descriptions should include the methods, models, and assumptions used as the bases for all values provided in SAR Section 12.2. A listing of isotope, quantity, form, and use of all required radiation sources containing byproduct, source, and special nuclear material exceeding 3.7 E+9 Bq (100 millicuries) that may warrant shielding design consideration, should be provided. SRP Section 12.3-12.4 indicates that the application should contain evidence of the methods to control personnel exposure from high dose rate components and that the staff will evaluate the dose rates in and around the spent fuel pool

areas.

As a result of the staff's review of the spent fuel source term and shielding information, the staff has the following questions.

1. While FSAR Table 12.2-9 provides the gamma radiation source for spent fuel, the application does not provide any information on the neutron source term. Please provide a neutron source term or discuss the neutron dose contribution if the neutron source term is negligible compared to the gamma source term.

2. FSAR Section 12.2.1.1.4 discusses the spent fuel source term for the entire core and FSAR Section 12.3.2.3 describes how the maximum gamma source term for the maximum fuel assembly is calculated. However, no information describing the spent fuel source term for fuel seated within the SFP is provided. Therefore, it is unclear how the minimum shield thicknesses around the spent fuel pool, in FSAR Table 12.3-4, were determined. Please update the FSAR to describe how the shielding around the spent fuel pool was determined. Ensure that shielding is adequate so that dose rates around the spent fuel pool will not exceed the dose to which they are zoned, during any possible spent fuel pool configuration allowed within the plant design.

3. FSAR Section 12.3.2.3 describes how shielding around the fuel transfer tube was determined. This is based on the 100 hour decayed spent fuel source strength provided in Table 12.2-9. Please provide a basis for using 100 hour decayed spent fuel for the shielding around the fuel transfer tube. If the application allows for fuel to be transferred earlier than 100 hours post-operation, please justify using 100 hours.

4. FSAR Figures 12.2-1 and 12.2-2 provide graphs of the dose rate from spent fuel vs. the axial and radial distances from a fuel assembly, however, the text of the application does not provide any information related to these graphs. Please update the text of FSAR Chapter 12 to provide information describing these figures. Include in the discussion how this information was developed, if this information is based on the maximum fuel assembly or an average assembly, if the dose rate includes the contributions from neutrons, and how it can be used to determine the maximum dose rate to an operator on the refueling platform during refueling operations and the maximum required shielding during transfer through the fuel transfer tube.

5. ANSI/ANS 57.1-1992, Section 6.3.4.1.5 indicates that fuel handling equipment shall be designed so that the operator will not be exposed to greater than 2.5 mrem/hour from an irradiated fuel unit, control component, or both, elevated to the up-position interlock with the pool at normal operating water level. The applicant references ANSI/ANS 57.1-1992 in FSAR Chapter 9.

    a. While FSAR Section 12.3.2.3 describes how shielding around the fuel transfer tube was determined, the application does not discuss the source term used to determine doses from a raised fuel assembly to an operator on the refueling machine and spent fuel handling machine, or the minimum amount of water shielding required over a raised fuel assembly. Please update the FSAR to discuss this source term and to provide a discussion of the maximum dose rate to an operator on the refueling platform including the minimum amount of water above a raised fuel assembly in order to maintain dose rates on the refueling platform within the ANSI/ANS 57.1-1992 criteria, or provide an

acceptable alternative.

b. FSAR Section 9.1.4.3, item d, under “The LLHS equipment has the following design features,” states that, “Mechanical stops in both the refueling machine and the spent fuel handling machine restrict withdrawal of the spent fuel assemblies. The resulting radiation level at a minimum water depth from the spent fuel is designed to meet the radiation dose limits in the work area when the shielding of the fuel handling equipment is taken into account.”

The Chapter 12 radiation zone figures indicate a dose rate above the spent fuel pool of less than 2.5 mrem/hour (0.025 mSv/hour), which is acceptable from for compliance with ANSI/ANS 57.1-1992, however, there are no radiation zone maps for refueling inside containment and the normal operation radiation zone maps provide a zone above the reactor cavity of less than 500 rem/hour (5000 mSv/hr). Please correct this apparent discrepancy. If the dose rate to an operator above the spent fuel pool and refueling cavity during fuel movement will be maintained less than 2.5 mrem/hour please ensure that it is clearly indicated in the FSAR, that the mechanical stops will prevent fuel from moving to a height that would result in dose rates of greater than 2.5 mrem/hour to an operator. If doses to an operator will not be maintained to less than 2.5 mrem/hour, please justify an alternative approach.

Response

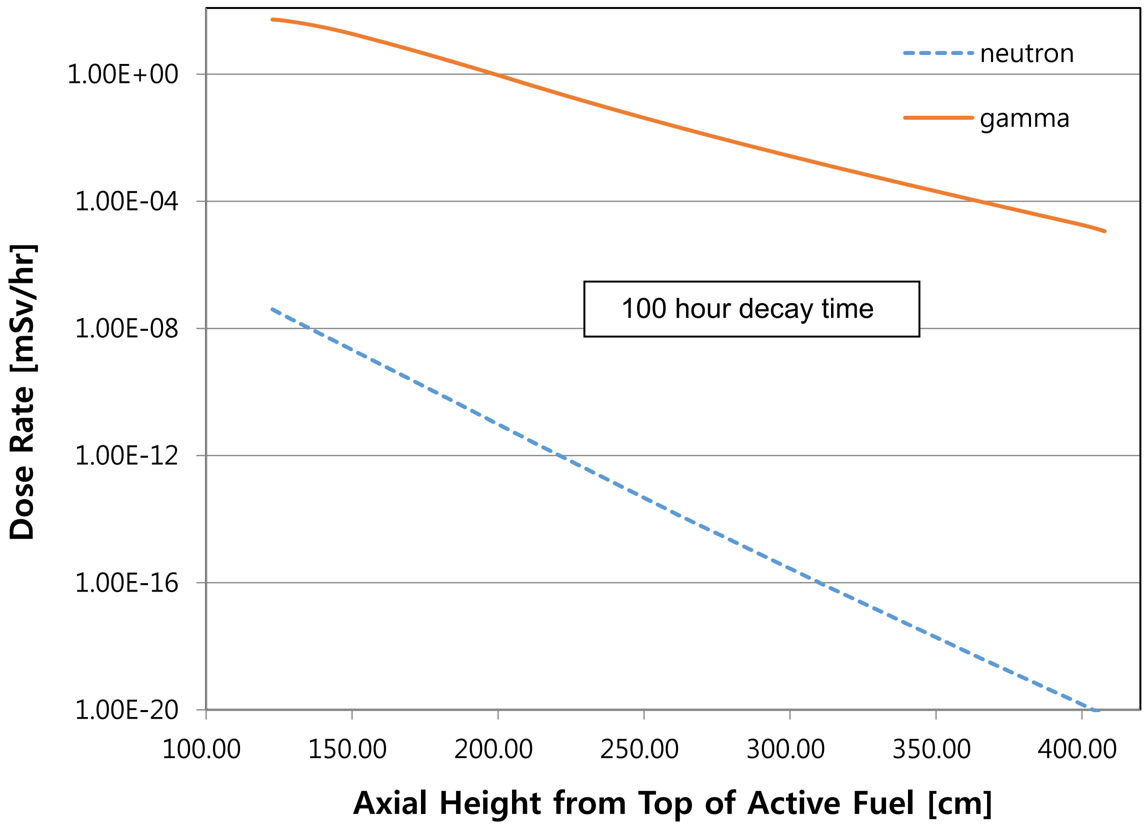
1. The neutron source term of a spent fuel assembly is calculated for 100 hours of decay time using the parameters listed in DCD Table 12.2-8. The initial enrichment of 4.2 w/o and 3 irradiation cycles with a discharge burnup of 56.4 GWD/MTU are applied to the neutron source term calculation. Table 1 shows the neutron and gamma source term for 100 hours of decay time after shutdown using the BUGLE-93 energy group. Dose rates due to the neutron and gamma source term in the refueling pool are calculated by using the DORT computer code. The dose rate in the refueling pool is provided in Figure 1. Figure 1 shows that the neutron source term in the spent fuel assembly contributes approximately 10^{-9} mSv/hr to the dose rate in the refueling pool. The gamma source term dose rate is several magnitudes higher than the neutron source term dose rate. Therefore, the neuron source term of spent fuel assembly is negligible compared to the gamma source term.

   DCD Sections 12.2.1.1.4 will be revised to address the dose rate contribution of neutron source term. Refer to Attachment 1 for the DCD markups.

   <Table 1> Spent Fuel Neutron and Gamma Source Term (100 hour after shutdown)

Energy Group	Neutron Energy (MeV)	Neutron /sec-FA
1	1.42E+01 - 1.73E+01	0.00E+00
2	1.22E+01 - 1.42E+01	1.54E+05
3	1.00E+01 - 1.22E+01	1.35E+06
4	8.61E+00 - 1.00E+01	2.08E+06
5	7.41E+00 - 8.61E+00	6.61E+06
6	6.07E+00 - 7.41E+00	2.00E+07
7	4.97E+00 - 6.07E+00	3.96E+07
8	3.68E+00 - 4.97E+00	1.11E+08
9	3.01E+00 - 3.68E+00	1.13E+08
10	2.73E+00 - 3.01E+00	5.82E+07
11	2.47E+00 - 2.73E+00	6.43E+07
12	2.37E+00 - 2.47E+00	2.71E+07
13	2.35E+00 - 2.37E+00	5.42E+06
14	2.23E+00 - 2.35E+00	3.13E+07
15	1.92E+00 - 2.23E+00	9.15E+07
16	1.65E+00 - 1.92E+00	8.77E+07
17	1.35E+00 - 1.65E+00	1.12E+08
18	1.00E+00 - 1.35E+00	1.49E+08
19	8.21E-01 - 1.00E+00	7.59E+07
20	7.43E-01 - 8.21E-01	3.62E+07
21	6.08E-01 - 7.43E-01	6.46E+07
22	4.98E-01 - 6.08E-01	5.21E+07
23	3.69E-01 - 4.98E-01	6.03E+07
24	2.97E-01 - 3.69E-01	3.13E+07
25	1.83E-01 - 2.97E-01	1.42E+04
26	1.11E-01 - 1.83E-01	8.95E+03
27	6.74E-02 - 1.11E-01	1.38E+03
28-47	1.00E-11 - 6.74E-02	0.00E+00

Energy Group	Gamma Energy (MeV)	Gamma /sec-FA
1	1.0E+01 - 1.4E+01	3.30E+04
2	8.0E+00 - 1.0E+01	6.97E+05
3	7.0E+00 - 8.0E+00	1.59E+06
4	6.0E+00 - 7.0E+00	4.74E+06
5	5.0E+00 - 6.0E+00	1.41E+07
6	4.0E+00 - 5.0E+00	4.17E+07
7	3.0E+00 - 4.0E+00	8.23E+12
8	2.0E+00 - 3.0E+00	2.54E+15
9	1.5E+00 - 2.0E+00	2.64E+16
10	1.0E+00 - 1.5E+00	9.46E+15
11	8.0E-01 - 1.0E+00	1.87E+16
12	7.0E-01 - 8.0E-01	7.84E+16
13	6.0E-01 - 7.0E-01	2.98E+16
14	4.0E-01 - 6.0E-01	6.06E+16
15	2.0E-01 - 4.0E-01	7.81E+16
16	1.0E-01 - 2.0E-01	1.00E+17
17	6.0E-02 - 1.0E-01	4.77E+16
18	3.0E-02 - 6.0E-02	5.43E+16
19	2.0E-02 - 3.0E-02	3.27E+16
20	1.0E-02 - 2.0E-02	1.18E+17



<Figure 1> Spent Fuel Assembly Dose Rate vs. Axial Distance in Refueling Pool

2. For the shielding design of the spent fuel pool, the source terms for both the spent fuel assemblies seated within the SFP and the water in the SFP are considered. The source terms for the spent fuel assemblies seated within the SFP are conservatively determined using the following design parameters and assumptions:

   - The number of spent fuel assemblies in the SFP is assumed to be 1,696, which is the maximum capacity of the SFP.

   - It is conservatively assumed that all spent fuel assemblies seated within the SFP have a decay time of 100 hours. The activities are calculated based on a reactor core power of 4,063 MW_t and a 3-cycle burnup of 56.4 GWD/MTU.

   - For the spent fuel assemblies in the storage rack, the radial power peaking factor is not considered.

   - Self-shielding effects of fuel assembly materials such as UO_2, Zircaloy-4 and H_2O are taken into account, while the shielding effects for the storage rack material and the stainless steel liner inside the pool are ignored for conservatism.

   For the source terms in the SFP water, the design basis source terms are based on the maximum SFP water specific activities presented in DCD Table 12.2-17. The dimensions of the SFP (width: 42 ft, length: 35.5 ft, height: 40 ft) are considered.

   By using this conservative approach in the calculations, it is confirmed that sufficient shielding is provided to ensure that the dose rates around the SFP will not exceed the limits for radiation zone designations during any possible SFP configuration allowed within the plant design.

   DCD Section 12.3.2.3 will be revised to address the shielding design of the spent fuel pool. Refer to Attachment 2 for the DCD markups.

3. Use of 100 hour decayed spent fuel for shielding design is determined based on the operating NPP experience in Korea. According to more than 40 reactor-years of operation for the four units of 1,000 MWe CE-type PWRs, the times to unload the first spent fuel assembly after reactor shutdown require longer than 100 hours. Tables 2 and 3 provide the actual times to unload the first spent fuel assembly after reactor shutdown for Hanbit Units 5 & 6 and Hanul Units 5 & 6, all of which are CE 1,000 MWe 2-loop PWRs. This duration typically includes RCS cooldown, disassembly of reactor ancillary equipment, disassembly of reactor head and other miscellaneous activities. Since the APR1400 will follow the same shutdown process, it is ensured that the minimum time of 100 hours for spent fuel decay time would not be exceeded.

<Table 2> Times to Unload Spent Fuel Assembly after Shutdown for Hanbit Units 5 & 6

Hanbit Unit 5		Hanbit Unit 6	
Overhaul Period	Time (hours)	Overhaul Period	Time (hours)
1st (3/17/2003 – 5/28/2003)	228	1st (11/19/2003 – 4/5/2004)	192
2nd (12/31/2003 – 5/28/2004)	286	2nd (1/18/2005 – 2/20/2005)	181
3rd (5/20/2005 – 6/26/2005)	203	3rd (2/28/2006 – 3/29/2006)	160
4th (11/12/2006 – 12/10/2006)	154	4th (6/16/2007 – 7/19/2007)	126
5th (4/4/2008 – 5/8/2008)	148	5th (11/5/2008 – 12/2/2008)	135
6th (8/23/2009 – 9/23/2009)	134	6th (3/18/2010 – 4/16/2010)	116
7th (12/22/2010 – 1/26/2011)	110	7th (6/20/2011 – 7/14/2011)	118
8th (4/30/2012 – 5/31/2012)	220	8th (11/6/2012 – 1/3/2013)	260
9th (12/12/2013 – 3/15/2014)	161	9th (5/16/2014 – 7/18/2014)	171

<Table 3> Times to Unload Spent Fuel Assembly after Shutdown for Hanul Units 5 & 6

Hanul Unit 5		Hanul Unit 6	
Overhaul Period	Time (hours)	Overhaul Period	Time (hours)
1st (6/20/2005 – 8/7/2005)	238	1st (3/25/2006 – 5/11/2006)	211
2nd (9/10/2006 – 10/13/2006)	188	2nd (5/31/2007 – 6/28/2007)	176
3rd (11/16/2007 – 12/12/2007)	169	3rd (9/6/2008 – 10/2/2008)	158
4th (4/11/2009 – 5/13/2009)	134	4th (2/16/2010 – 3/17/2010)	125
5th (9/18/2010 – 10/9/2010)	122	5th (6/8/2011 – 7/1/2011)	114
6th (11/24/2011 – 12/21/2011)	121	6th (10/25/2012 – 12/6/2012)	150
7th (5/3/2013 – 6/16/2013)	142	7th (4/7/2014 – 5/17/2014)	166
8th (11/14/2014 – 1/1/2015)	174		

4. The dose rates from a spent fuel assembly are calculated by using the 2-D DORT computer code using the gamma source terms provided in DCD Table 12.2-9. The gamma source term is calculated based on the maximum fuel assembly. The neutron source term is not included in this calculation because the dose contribution due to neutrons is negligible. The macroscopic cross sections are obtained using the GIP computer code. The flux-to-dose conversion factors in ANSI/ANS-6.1.1-1991 are used. The R-Z DORT model includes the active fuel assembly and fuel handling apparatuses. The axial gamma source distribution is assumed to be the long term axial power distribution and the radial source distribution is assumed to be flat.

   DCD Figure 12.2-1 shows the dose rates for the refueling machine operator during refueling operation according to the axial height from top of active fuel in the refueling pool. “Axial height from top of active fuel” is determined by the refueling machine hoisting limit (mechanical stop) of the fuel assembly during refueling operation.

   DCD Figure 12.2-2 shows the dose rates according to the radial distance from edge of active fuel in the refueling pool.

   The dose rate to an operator on the refueling platform is determined by the dose rate value corresponding to the axial height (or water depth) from top of active fuel to the refueling pool water level after 100 hour decay time as provided in the DCD Figure 12.2-1.

   DCD Sections 12.2.1.1.4 and 12.2.5 will be revised to provide the information describing Figures 12.2-1 and 12.2-2. Refer to Attachment 1 for the DCD markups.

   In determining the shielding design requirements for the fuel transfer tube, DCD Figures 12.2-1 and 12.2-2 are not used. The graphs provided in the figures present the dose rates versus water depth, while the fuel transfer tube shielding calculations require additional modeling for geometry and materials. Therefore, the fuel transfer tube shielding calculations make use of maximum shielding source terms provided in DCD Table 12.2-9. This is currently described in the third paragraph of Page 12.3-28 in DCD Section 12.3.2.3.

5. a. The transport calculations using DORT code are performed to obtain the dose rates from a spent fuel assembly in the refueling pool and the spent fuel pool at several decay times after shutdown. The BUGLE-93 cross section library is used for the gamma transport calculation. The decay gamma sources from a spent fuel assembly are taken from depletion and decay calculations of a PLUS7 fuel assembly for APR1400 operating conditions. The initial enrichment of 4.2 w/o and 3 cycle operations with discharge burnup of 56.4 GWD/MTU are applied for the decay gamma source calculations. The decay gamma sources for 100 hour decay time are shown in Table 1 of response number 1.

   The dose rate to an operator on the refueling platform is determined by the dose rate value corresponding to the axial distance (or water depth) from top of active fuel to the refueling pool water level after 100 hour decay, as provided in DCD Figure 12.2-1. Since mechanical stops in both the refueling machine and spent fuel handling machine restrict withdrawal of the spent fuel assemblies above the minimum safe water cover depth of 274.32 cm (9 ft) from top of active fuel, an operator on the refueling platform will not be exposed to the radiation dose limits (0.025 mSv/hr) in the work area when the shielding of the fuel handling equipment is taken into account.

   DCD Section 12.3.2.3 will be revised to address the shielding design of a raised spent fuel assembly. Refer to Attachment 2 for the DCD markups.

   b. The containment areas above Elevation 156’-0” will be maintained with a dose rate that is less than 2.5 mrem/hr during refueling operation. For clarity, the radiation zone maps for these areas will be revised to include a note that states “This area is Zone 2 during refueling operation”. Refer to Attachment 3 for the DCD markups.

Impact on DCD

DCD Section 12.2.1.1.4 and Section 12.2.5 will be revised as indicated in the Attachment 1.

DCD Section 12.3.2.3 will be revised as indicated in the Attachment 2.

DCD Figures 12.3-7 and 12.3-8 will be updated to include a note as indicated in Attachment 3.

Impact on PRA

There is no impact on the PRA.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical and Environmental Reports.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

The neutron source term is not included because the dose rate contribution of neutron source term is negligible. .

assumed. No credit is taken for the operation of the condensate polisher demineralizers. Assumptions for calculation of main steam system (MSS) radiation sources are presented in Table 11.1-5, and results of the calculation are given in Table 12.2-18.

12.2.1.1.4 Spent Fuel Handling and Transfer

The spent fuel assemblies are the predominant source of radiation in the reactor containment building (RCB) after plant shutdown for refueling. A reactor operating time to reach a near-equilibrium buildup level of fission products based on the rated power operation is used in determining the source strengths. The parameters used in the spent fuel decay gamma source calculation, such as fuel enrichment, specific core power, and discharge burnup, are given in Table 12.2-8. The spent fuel decay gamma source is given in Table 12.2-9. Fuel assembly dose rates as a function of distance in the refueling pool water and time after shutdown are shown in Figures 12.2-1 and 12.2-2.

12.2.1.1.5 Processing Systems

12.2.1.1.5.1 Chemical and Volume Control System

The shielding design for chemical and volume control system (CVCS) components is based on the maximum expected radioactivity in each of the components. The component source terms are given in Tables 12.2-10 through 12.2-14.

a. Heat exchangers

The maximum values for the CVCS heat exchanger (HX) radionuclide inventories are presented in Table 12.2-10.

The total radioactivity inventory for the CVCS heat exchangers is based on the tube-side water volume (including the shell-side water volume for regenerative heat exchangers).

The dose rates in the refueling pool are calculated by using neutron transport code, DORT (Reference 7), and flux-to-dose factors in ANSI/ANS-6.1.1 (Reference 8). These figures are based on the gamma source terms in Table 12.2-9. These figures provide the information of dose rates to an operator during refueling operation.

12.2.5 References

1. NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," Rev. 1, U.S. Nuclear Regulatory Commission, April 1985.

2. DIJESTER, "A Program to Compute Radioactive Decay in Fluid Flow Systems," Program No. 9.8. 060-1.0, D. J. Pichurski, April 1976.

3. 10 CFR Part Part 20, Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," U.S. Nuclear Regulatory Commission.

4. NUREG-0737, Item II.B.2.3, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Post-Accident Operations," U.S. Nuclear Regulatory Commission, 1980.

5. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," U.S. Nuclear Regulatory Commission, July 2000.

6. ANSI/ANS-18.1-1999, "Radioactive Source Term for Normal Operation of Light Water Reactors," American Nuclear Society, 1999.

7. CCC-650/DOORS3.2, "One-, Two- and Three- Dimensional Discrete Ordinates Neutron/Photon Transport Code System," Radiation Safety Information Computational Center, ORNL, 1998.

8. ANSI/ANS-6.1.1, "American National Standard for Neutron and Gamma-Ray Fluence-to-Dose Factors," American Nuclear Society, August 1991.

provided for all accessible plant areas capable of radiation levels greater than 1 Gy/hr. Areas with the potential for radiation greater than 1 Gy/hr are listed in Table 12.3-5.

Transient sources greater than 1 Gy/hr are considered in the shielding design to provide reasonable assurance that adequate shielding is provided. One such source is a spent fuel assembly. During transfer of a spent fuel assembly through the fuel transfer tube, adjacent areas may have elevated radiation levels. Streaming from this source up through the joint between the reactor containment building and the auxiliary building has been a concern for the current generation of nuclear plants. The APR1400 design uses connected building structures to reduce the potential for streaming. In addition, sufficient concrete shielding is provided to maintain radiation levels in adjacent areas ALARA during spent fuel transfer. This permits personnel to perform maintenance and inspection activities in a lower-radiation area and reduces the potential for high-radiation levels adversely affecting refueling outage schedules. An inspection area is provided for the fuel transfer tube. Access control is provided by the personnel airlock through the reactor containment building.

Sufficient shielding provides reasonable assurance that the areas adjacent to the spent fuel transfer tube are accessible and expected radiation zones are consistent with those in Figure 12.3-52 during transfer of a spent fuel assembly. The shielding design of the fuel transfer tube is based on the 100 hr decayed spent fuel source strengths provided in Table 12.2-9. The gamma source strengths given in units of [MeV/W-sec] are converted to [photons/sec] by multiplying the gamma source strength values by the thermal power per fuel assembly in [W] and dividing by the source energy in [MeV]. Then, the shielding source term is determined by multiplying this calculated value by the radial power peaking factor of 1.55 and by the number of fuel assemblies transferred through the transfer tube, which is two (2).

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Typically, pipe chases do not need to be accessed frequently. The APR1400 design minimizes locating components such as valves in pipe chases to minimize plant personnel access to pipe chases and to reduce the potential for radiation exposure. When access is needed, radiation protection personnel conduct a survey of the area to determine the strength and location of radiation sources within the pipe chase. Temporary shielding is used to minimize personnel exposure. If the primary source of radiation in the pipe chase is spent resin or slurry transfer piping, precautions are taken by operating personnel to provide reasonable assurance that no spent resin is transferred while personnel are in the

"A"

The spent fuel pool is designed to comply with the following shielding design criteria : (a) to provide sufficient shielding so as to meet the radiation zoning requirements adjacent to the SFP and (b) to maintain the dose rate to the spent fuel handling operator less than 0.025 mSv/hr as specified in ANSI/ANS-57.1-1992.

In determining the shielding requirements around the SFP, the source terms for the spent fuel assemblies seated in the SFP storage racks and for the SFP water are considered. To add conservatism to the design, the SFP is assumed to be filled with 1,696 spent fuel assemblies decayed for 100 hours, for which the source term is provided in Table 12.2-9. The radial power peaking factor is not considered. The source region is assumed to be homogeneously mixed with UO_2, Zircalory-4 and H_2O in the shielding calculations, while the shielding effects for the SFP storage racks and the SFP liner are not taken into account. For the source terms in the SFP water, the design basis specific activities provided in Table 12.2-17 are used. By using this conservative approach in the shielding calculations, it is ensured that the minimum shielding requirements around the SFP provided in Table 12.3-4 meet the radiation zoning designations presented in Figures 12.3-3 through 12.3-5.

The dose rate to an operator on the refueling platform is determined by the dose rate value corresponding to the axial distance (or water depth) from top of active fuel to the refueling pool water level after 100 hour decay in Figure 12.2-1. Since mechanical stop in both the refueling machine and spent fuel handling machine restrict withdrawal of the spent fuel assemblies above the minimum safe water cover depth of 274.32 cm (9 ft) from top of active fuel, an operator on the refueling platform will not be exposed to the radiation dose limits in the work area when the shielding of the fuel handling equipment is taken into account.

~~**Security-Related Information – Withhold Under 10 CFR 2.390**~~

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