# 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. :** **23-7929**

**SRP Section:** **14.3-INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA**

**Application Section: 12.2**

**Date of RAI Issue:** **06/08/2015**

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## Question No.12.02-6

1. Please identify all equipment and areas within the refueling cavity which are intended to be used or could be used to store and handle fuel.

2. Indicate if the current design of the APR1400 physically excludes the intermediate fuel storage rack discussed in the September 2013 application.

3. SRP Section 14.3 states that the type of information and level of detail in Tier 1 are based on a graded approach commensurate with the safety significance of the structures, systems, and components for the design. Staff believes that information regarding which equipment stores and handles fuel is safety significant information which should be included in Tier 1. Therefore, please update Tier 1, Section 2.7.4 to identify all equipment and locations within the plant which will be used to handle or store reactor fuel. Include a statement in Tier 1 indicating that no other equipment or locations will be used to store or handle fuel beyond the items listed.

## Response

1. Tier 1 Table 2.7.4.4-1 will be revised to indicate all equipment and locations where fuel will be handled or stored.

2. APR 1400 performs the full core offload during refueling operation, not considering fuel shuffling, so Intermediate Fuel Storage Racks for the temporary storage of fuel in the refueling pool are not required.

However, to store cut-up pieces of spent Control Element Assemblies and In Core Instrumentations, the CEA/ICI Transport Container positioned in the Upper Guide Structure Laydown Area should be moved to the Core Support Barrel Laydown Area in advance of where the Refueling Machine is allowed to be handled in order for the container to be transferred to the Spent Fuel Storage Pool in the Auxiliary Building.

Therefore, the Intermediate Transport Container Storage Rack within the Core Support Barrel Laydown Area is installed to store the CEA/ICI Transport Container temporarily.

3. No other equipment or locations will be used to store or handle fuel beyond the items listed Table 2.7.4.4-1.

   Tier 1 Table 2.7.4.4-1 will be revised.

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**Impact on DCD**

Tier 1 Table 2.7.4.4-1 will be revised as indicated on the attached markup.

**Impact on PRA**

There is no impact on the PRA.

**Impact on Technical Specifications**

There is no impact on Technical Specifications.

**Impact on Technical/Topical/Environmental Reports**

There is no impact on any Technical/Topical/Environmental Reports.

Table 2.7.4.4-1

Light Load Handling System Equipment Location/Characteristics

| Equipment Name | Location | ASME Section III Class | Seismic Category | Fuel Handling |
|---|---|---|---|---|
| Refueling Machine | ~~Reactor~~ Containment Building | NNS | II | Yes |
| CEA Change platform | ~~Reactor~~ Containment Building | NNS | II | No |
| CEA Elevator | ~~Reactor~~ Containment Building | NNS | II | No |
| Spent Fuel Handling Machine | Auxiliary Building | NNS | II | Yes |
| New Fuel Elevator | Auxiliary Building | NNS | II | Yes |
| Fuel Handling hoist of overhead crane | Auxiliary Building | NNS | II | Yes |
| Double blind flange assembly | ~~Reactor~~ Containment Building | 2 | I | No |

Deleted

Added

No other equipment or locations will be used to store or handle fuel beyond the items listed Table 2.7.4.4-1.

Added

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**Docket No. 52-046**

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|---|---|
| **RAI No.:** | **23-7929** |
| **SRP Section:** | **12.02 – Radiation Sources** |
| **Application Section:** | **12.2** |
| **Date of RAI Issue:** | **06/08/2015** |

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## Question No. 12.02-7

REQUIREMENTS

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.

SRP Section 12.2 indicates that the description of airborne sources should include a tabulation of the calculated concentrations of radioactive material, by nuclide, expected during normal operation, AOOs, and accident conditions for areas normally occupied by operating personnel and that the FSAR should provide the models and parameters used for the calculations.

ISSUE

FSAR Section 12.2.2.3 describes how the applicant determined airborne activity concentrations throughout the plant. This section provides an equation for calculating the equilibrium airborne calculations in rooms and cubicles. In addition, Table 12.2-26 provides assumptions and parameters used in the airborne source term calculations.

Staff is unable to duplicate the applicant's airborne source terms due to a lack of information and apparent inconsistencies between Chapter 12 and other FSAR questions. Therefore, the staff has the following questions related to airborne source terms.

INFORMATION NEEDED

1. The equation in FSAR Section 12.2.2.3 for calculating equilibrium airborne concentration contains a "P" variable for the fraction of activity released to air. The applicant does not provide any information on what the P values are or how they were

determined. Therefore, the applicant should describe how the P values were determined and list the P values in FSAR Table 12.2-26.

2. FSAR Table 12.2-26 provides parameters for calculating airborne source terms. This table indicates that the RCS letdown flow rate is “364 L/min (80 gpm).” However, FSAR Table 9.3.4-3, which provides parameters for the chemical and volume control system, indicates that the normal RCS letdown and purification flow rate is “302.8 L/min (80 gpm).” Please update the FSAR to ensure both tables contain the correct values. If an incorrect value was used in any calculations or assumptions made in the FSAR (in airborne activity calculations or otherwise) please make the appropriate modifications and update the FSAR as appropriate, or justify an alternative.

3. As indicated above, Table 12.2-26 provides assumptions and parameters for the airborne source term calculations. However, some of the parameters given would appear to provide extra information beyond what it needed to perform the calculation. For example, Table 12.2-26 (1 of 8) provides a filter efficiency for low-volume purge filters for calculating containment building airborne activity, even though it does not appear the efficiency of these filters would have an effect on the airborne source term in containment since air sent through the purge system is not returned to containment.

   It is acceptable to provide additional information beyond the parameters needed to perform the calculation, however, staff cannot duplicate the applicant’s results for airborne source terms.

   a. Therefore, the staff requests that the applicant select one non-noble gas isotope and demonstrate how the values in Table 12.2-23 (1 of 4) “Reactor Containment Building (Normal Operation)” and Table 12.2-23 (2 of 4) “Reactor Containment Building (48 hr after Shutdown)” for that isotope were obtained.

   b. In addition, select one cubicle in FSAR Table 12.2-23 (3 of 4) “Auxiliary Building Cubicles (Normal Operation)” (for example, Charging Pump room) and demonstrate how all values for “Airborne Radioactivity Concentration” and “Derived Air Concentration (DAC) Fraction” were obtained.

4. In sheets 3 of 4 and 4 of 4 of FSAR Table 12.2-23, the applicant contains column providing concentrations for Kr and Xe together and Br and I together and derived air concentration fractions for each, however the combined concentration of many different isotopes does not appear to have much meaning and it is unclear how it is compared to the DAC values when DAC values are isotope specific. Please explain what the values in the following columns represent; 1) “Kr, Xe” under Airborne Activity concentration; 2) “Br, I” under Airborne Activity Concentration; 3) “Kr, Xe” under DAC fraction; 4) “Br, I” under DAC fraction; and 5) Total under DAC Fraction.

5. In sheets 3 of 4 and 4 of 4 of FSAR Table 12.2-23, please indicate if other radionuclides beyond those shown in the tables were considered and how the specific nuclides shown were selected for inclusion in the table.

6. Please indicate why the Kr, Xe and Br, I airborne activity concentrations for the fuel handling area (normal operations) and fuel handling area (refueling) in Table 12.2-23, sheet 3 of 4 are blank when all cubicles listed in sheets 3 of 4 and 4 of 4 of FSAR Table 12.2-23 contain numerical values in those columns (even if the value is zero). Please update FSAR Table 12.2-23 sheet 3 of 4 to provide numerical values in these columns. In addition, provide justification for the values that are provided.

7. Please demonstrate how the information in FSAR Table 12.2-26 (2 of 8) was used to develop the airborne activity source term for the fuel handling area (both normal and refueling). In the response please indicate if the information in FSAR Table 12.2-17, “Fission and Corrosion Product Activities in the Spent Fuel Pool” was used in calculating the airborne activity concentrations in the fuel handling area (both normal and refueling).

**Response**

1. “P” variable indicates the partition factor or the fraction of the leaked activity that becomes airborne. The values used to determine the airborne activity are dependent upon the types of nuclide and temperature of the leaking fluid and are as follows:

   - 1.0 for noble gases
   - $1 \times 10^{-3}$ for halogens in a cold liquid (< 120 °F) and 0.1 for halogens in a hot liquid (> 120 °F)
   - 0.53 for H-3 in primary coolant and 0.1 for H-3 in cold liquids
   - 0.0 for all other nuclides

   Basis of the “P” values are described below:

   - All noble gases (i.e., Xe and Kr) in the leaked fluid are assumed to be suspended into the air conservatively, because the noble gases have extremely low chemical reactivity.
   - The partition factors for halogens (i.e., I and Br) in the leaked fluid with greater than 120 °F and less than 120 °F are conservatively selected as 0.1 and 0.001, respectively. Regulatory Guide (RG) 1.183, which provides acceptable radiological analysis assumptions for use in conjunction with the accident source terms, specifies that the partition factor for elemental iodine in the bulk water in the steam generators (SG) may be assumed to 0.01. This partition factor is derived based on

the condition that the bulk water in SGs is at the boiling condition of high temperature and pressure. This value is a maximum value of the range from 0.00005 to 0.01 presented in NUREG-0409, “Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident”. Also, according to a European Commission Report, “Realistic Methods for Calculating the Release of Radioactivity Following SG Tube Rupture Faults, EUR 15615, 1994”, under recirculating condition, a partition factor value of $10^{-5}$ is recommended. Therefore, the partition factors of 0.1 and 0.001 used for estimating airborne source terms in FSAR Section 12.2.2.3 are considered conservative because the temperature and pressure conditions in the leaked fluid are lower than those in the SG.

- For H-3, the fraction of airborne radioactivity that can be transported through air is derived based on the flashing fraction (ff), which means the portion of discharged fluids that flashes to vapor. This value is calculated based on the enthalpy difference under circumstance of coolant leakage by assuming the leakage to be a constant enthalpy process and expressed as follows in accordance with RG 1.183:

$$ff = \frac{h_{rcs} - h_f}{h_{fg}}$$

- In the cases that the temperature of the leakage is less than 100°C (212 °F) or the calculated flashing fraction is less than 10%, the amount of radionuclides that becomes airborne is assumed to be 10% of the total iodine activity in the leaked fluid.

- The suspension of particulate nuclides is limited by the moisture carryover, of which the phenomenon is led by droplet formation of steam due to boiling, or by atomization of water leaked from the components in a system, of which fineness is dependent upon the pressure drop across the leak path. Therefore, the amount of airborne nuclides can be considered negligible so that no particulate nuclide in the leaked fluid is assumed to be suspended into the air.

2. The RCS letdown flow rate of 364 L/min (80 gpm) in Table 12.2-26 is an editorial error. It will be updated to 302.8 L/min (80 gpm) as indicated in Attachment.

3. Table 12.2-26 (1 of 8) provides the filter efficiency for low volume purge system. This value is not used for calculating the airborne activity in the containment during normal operation. Instead, it is used to calculate the airborne source term in containment after shutdown.

   The containment low volume purge system is designed either to return the processed air to the containment or to exhaust directly to the environment. In the calculation of airborne activity during shutdown operation, a recirculation fraction of 100% is assumed

to obtain the conservative results. Therefore, the whole volume of air that passes through the filter of the purge system is assumed to flow back into the containment.

a. I-131 is selected to explain how the resultant values were obtained. The source term calculation for containment building airborne activity is based on 0.25% fuel defect. The RCS I-131 specific activity corresponding to this fuel defect is 2.48E+04 Bq/g. The leak rate of RCS is assumed to be 0.5 gpm, which is the LCO for unidentified leakage defined in the Technical Specifications. The flashed portion of the coolant leakage becomes the airborne, and the flashing fraction is calculated considering the difference of enthalpy between the inside and the outside of the RCS. The flashing fraction of 0.5687 is used in the calculation for normal operation. This value is determined by using the formula given in response to the above Question No. 1. During normal operation, temperature of the RCS is 613.4 ˚F and that of the containment building atmosphere is 68 ˚F.

   The airborne concentrations during normal operation provided in Table 12.2-23 (1 of 4) are the concentrations at 100 hours before shutdown. Also, as indicated in Table 12.2-26 (1 of 8), the low volume purge system is assumed to start operating at 100 hours before shutdown, which means that the filter efficiency of the purge system is not used for this case. Therefore, I-131 decay constant of 3.59E-03/hr is the only removal mechanism during normal operation. In addition, normal operation time of 18 months is assumed in the calculation. As a result, the airborne activity in the containment during normal operation can be determined by using the parameters indicated above and the other parameters in Table 12.2-26 (1 of 8).

   In case of the refueling condition, the airborne concentration at 48 hours after shutdown is obtained as listed in Table 12.2-23 (2 of 4). And, the filter efficiency of 99% (except for noble gas and tritium) is considered based on the conservative assumption that the air is sent to the low volume purge system filters and flows back into the containment. Therefore, the effective removal rate is a combination of this filter efficiency and the decay constant of I-131. After shutdown, the temperature of the reactor coolant is assumed to decrease linearly depending on the limiting temperature of plant operating modes (i.e., Modes 1 to 6). Based on the temperature changes, time-dependent flashing fraction can be calculated. And, the RCS leak rate of 0.5 gpm is assumed to decrease proportionally to the square root of the gap between two pressures of reactor coolant system and containment building atmosphere. The airborne activity in the containment at 48 hours after the shutdown is then calculated using a set of first-order differential equations and the above design parameters and assumptions.

b. The charging pump room (055-A42A) is selected to illustrate how to obtain the airborne concentration and the DAC fraction,

To calculate the air borne concentration in the charging pump room the following equation is used:

$$C_A = \frac{C\ L\ P}{7.48\ (\lambda\ V + F)}$$

where,

$C_A$ = the airborne concentration in the charging pump room ($Bq/cm^3$)

$C$ = liquid or gas concentration in charging pump ($Bq/cm^3$)

$L$ = leak rate (gpm)

$P$ = fraction of the leaked activity that becomes airborne

$\lambda$ = decay constant ($min^{-1}$)

$V$ = enclosed room volume ($ft^3$)

$F$ = air exhaust flow rate (cfm)

7.48 = conversion factor (7.48 $gal/ft^3$).

Since the charging pumps are directly connected to the volume control tank (VCT), the concentrations, C, of halogen nuclides in the charging pump are obtained from those in the VCT and the concentrations of noble gases are from the primary coolant.

Leak rates (L) of the components in the charging pump room are determined using the maximum allowable design leak rates of the components shown in Table 1.

**Table 1. Maximum Allowable Leak Rates for Each Component**

| Component | Leakage Assumptions |
|---|---|
| Valves | |
| Disk Leakage | 10 $cm^3$/hr/inch seat diameter |
| Stem Leakage | 10 $cm^3$/hr/inch seat diameter |
| Pumps | |
| Centrifugal (mechanical seal) (Except SI Pumps) | • 50 $cm^3$/hr/seal during normal operating conditions with availability of seal cooling water<br>• 100 $cm^3$/hr/seal during loss of externally supplied cooling water |
| Positive Displacement | 1 gal/hr |
| SI Pumps | 1,000 $cm^3$/hr/seal (each pump) |
| Flanges | 30 $cm^3$/hr |

According to the detailed layout of the reference APR1400, there are one(1) pump, six (6) flanges, one(1) 1-inch diameter valve, two (2) 4-inch diameter valves and four (4) 3-inch diameter valves in the charging pump room. Therefore, the total leak rate is determined using the number of components and the maximum allowable leak rates of the corresponding components.

The partition factors (P) used in the calculation is as follows:

- 1 for noble gas
- $1 \times 10^{-3}$ for halogen
- 0.53 for H-3

These values are discussed in response to the above Question No. 1.

The volume of the charging pump room (V) is obtained from the general arrangement of the APR1400 layout and is calculated to be 9,380 $ft^3$ (≈17 ft (W) × 30 ft (L) × 23 ft (H)× 0.8 (Free volume fraction)) .

The required air flow rate of the charging pump room which maintains the sum of the DAC fractions less than 1.0 is calculated to be 361 cfm. In order to apply ALARA principle, the air flow rate (F) is determined to be 500 cfm, which makes the sum of DAC fractions to be 0.729 as presented in Table 12.2-23 (3 of 4). Table 2 shows the nuclide-specific airborne concentrations and their DAC fraction values for the charging pump room. In Table 2, the sum nuclide-specific DAC fraction ($F_{DAC}$) is calculated using the following equation:

$$\mathrm{F_{DAC}} = \sum_{\mathrm{i}} \frac{\mathrm{C_{Ai}}}{\mathrm{DAC_i}}$$

where,

$C_{Ai}$ = the airborne concentration of each isotope

$DAC_i$ = Derived Air Concentration of each isotope ($Bq/cm^3$) in 10 CFR 20 Appendix B

**Table 2. Nuclide-specific Airborne Concentrations and DAC Fractions with Air Flow Rate of 500 cfm in Charging Pump Room**

| Nuclide | $C_A$(Bq/cm$^3$) | $F_{DAC}$ |
|---|---|---|
| Kr-85m | 5.13E-03 | 6.93E-03 |
| Kr-85 | 2.30E-02 | 6.21E-03 |
| Kr-87 | 3.60E-03 | 1.95E-02 |
| Kr-88 | 1.09E-02 | 1.47E-01 |
| Xe-131m | 2.30E-02 | 1.55E-03 |
| Xe-133m | 1.39E-03 | 3.76E-04 |
| Xe-133 | 1.49E+00 | 4.04E-01 |
| Xe-135m | 1.67E-03 | 5.01E-03 |
| Xe-135 | 2.99E-02 | 8.09E-02 |
| Xe-138 | 1.52E-03 | 1.03E-02 |
| Br-84 | 3.77E-10 | 5.09E-10 |
| I-131 | 7.95E-08 | 1.07E-04 |
| I-132 | 1.85E-08 | 1.66E-07 |
| I-133 | 1.11E-07 | 3.00E-05 |
| I-134 | 9.33E-09 | 1.26E-08 |
| I-135 | 6.13E-08 | 2.37E-06 |
| H-3 | 3.56E-02 | 4.82E-02 |
| Sum | - | 7.29E-01 |

4. In order to avoid complexity of the DCD tables related to airborne concentrations, the isotopes considered in the airborne concentration calculations are divided into three groups of “Kr, Xe”, “Br, I” and “H-3”. Isotopes included in each of the group are as follows:

    - Kr, Xe : Noble gases
        - Kr-85m, Kr-85, Kr-87, Kr-88
        - Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135
    - Br, I : Halogens
        - Br-84
        - I-131, I-132, I-133, I-134, I-135
    - H-3 : Tritium

    Therefore, the airborne radioactivity concentration given in the units of Bq/cm$^3$ in column “Kr, Xe” in Table 12.2-23 is the sum of concentrations of the four (4) Krypton

and five (5) Xenon isotopes. Similarly, the column labeled as “Br, I” presents the sum of concentrations of one (1) Bromine and five (5) Iodine isotopes. For “H-3”, there is only one isotope in this group. In addition, since the values in the “DAC Fraction” columns represent the sum of the DAC fractions of all isotopes in each group, of which DAC fractions for each isotope is calculated separately, the DAC fraction values in Table 12.2-23 are not directly obtained from the concentration values for each isotope group. For example the values for “Charging Pump Room” in Table 12.2-23 are obtained as presented in Table 2 of the above response to Question No. 3.

5. As discussed in the response to above Question No. 4, the isotopes considered in the airborne activity concentrations are noble gases, halogens and tritium. These are the most important radionuclides to be considered. Particulates that might exist in the leaking liquid are assumed to be dissolved in the liquid without any partitioning (i.e., partition factor of 0.0) as indicated in the response to Question No. 1.

6. The airborne activity in the fuel handling area is calculated by only considering tritium. It is because tritium was considered as the most dominant nuclide for airborne activity in fuel handling area. Since the noble gases of the RCS are degassed by CVCS gas stripper during shutdown cooling operations or released to the Containment Building while the reactor head is open, these nuclides do not exist in the spent fuel pool. For iodines and particulates, the release from the spent fuel pool is normally considered negligible compared to the rates of tritium evaporation.

7. Tritium airborne activity in the fuel handling area is calculated using the design parameters given in Table 12.2-26 (2 of 8). Since the tritium activity is determined by the evaporation rate of the spent fuel pool (SFP), most of the parameters such as air pressure, temperature, relative humidity, surface area of SFP, and wind speed are used to calculate the evaporation rate of the SFP. The surface area changes according to the operating mode, because the SFP water is mixed with reactor coolant, reactor makeup water, and in-containment refueling water during refueling. For normal operation, the evaporation is assumed to occur only in SFP. During refueling operation, the evaporation could also occur in refueling pool and canal. This difference of the surface area is conservatively reflected to the calculation for refueling case.

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**Impact on DCD**

DCD section Table 12.2-26 will be updated as indicated in the Attachment.

**Impact on PRA**

There is no impact on the PRA.

**Impact on Technical Specifications**

There is no impact on the Technical Specifications.

**Impact on Technical/Topical/Environmental Reports**

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

Table 12.2-26 (1 of 8)

Assumptions and Parameters Used in Airborne Source Term Calculations

302.8

Reactor Containment Building

| Assumptions/Parameters | Values |
|---|---|
| Fuel failure | 0.25 % fuel defect |
| RCS leak rate | 1.89 L/min (0.5 gpm)<br>(design basis leak rate) |
| RCS letdown flow | ~~364~~ L/min (80 gpm) |
| Iodine spike | Not considered |
| Low-volume purge flow rate | 2.549E+03 $m^3$/hr (1,500 cfm)<br>(Assumed that CLVPS operates for 100 hours before shutdown) |
| High-volume purge flow rate | 9.14E+04 $m^3$/hr (54,000 cfm)<br>(Assumes that CHVPS operates at cold shutdown operation mode for 29 hours) |
| Filter efficiency for low-volume purge filters | Particulate: 99 %<br>Halogen: 99 % |
| Gas stripping | No gas stripping |
| Containment free volume | 8.858E+04 $m^3$ (3.128E+06 $ft^3$) |

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## Question No. 12.02-8

REQUIREMENT

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.

INFORMATION NEEDED

FSAR Table 12.2-25 provides cask loading pit source dimensions that appear consistent with the dimensions of a single fuel assembly. 1) Please indicate if spent fuel is intended to be located within the cask load pit other than during cask loading operations. 2) Please update the FSAR to specify which source term was used to determine cask loading pit shielding.

## Response

FSAR Table 12.2-25 (3 of 3) provides the source dimensions for a single fuel assembly used for cask loading pit shielding. These dimensions are used to determine the shielding requirements of the cask loading pit assuming that a spent fuel assembly is being loaded into the cask in the cask loading pit.

The cask loading pit is designed only to load the spent fuel into the cask for transport to a storage area. Other than this cask loading operation, the cask loading pit remains empty.

The 1,000-hour decayed source term for a spent fuel assembly given in FSAR Table 12.2-9 is used to conservatively determine the cask loading pit shielding. The FSAR will be updated to specify this source term is used to determine cask loading pit shielding.

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**Impact on DCD**

DCD section 12.3.2.3 will be updated as indicated in the Attachment.

**Impact on PRA**

There is no impact on the PRA.

**Impact on Technical Specifications**

There is no impact on the Technical Specifications.

**Impact on Technical/Topical/Environmental Reports**

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

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).

For shielding of the cask loading pit, it is assumed that a single spent fuel assembly is being loaded into the cask within the cask loading pit. The source term of the spent fuel assembly is conservatively assumed to be decayed for 1,000 hours after reactor shutdown. This source term is also provided in Table 12.2-9 and the gamma source strengths used for the cask loading pit shielding are derived using the same methods as those for the fuel transfer tube shielding except that the number of fuel assemblies is one.

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

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## Question No. 12.02-9

REQUIREMENTS

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.

INFORMATION REQUESTED

FSAR Table 12.2-25 provides the material and densities of radiation sources. For the shutdown cooling heat exchanger (SC HX) the applicant indicates that the material is 6% vapor, and that the density contribution for the vapor phase is 0.453 grams per cubic centimeter. This density appears inconsistent for what would be expected for the density of water vapor or air and appears inconsistent with the density provided for other equipment containing vapor. Please correct this apparent discrepancy or provide justification for the 0.453 grams per cubic centimeter vapor density value.

## Response

In order to estimate the radiation sources of SC HX, it is assumed that the tube-side water, shell-side water and tube wall material are homogeneously mixed in the source region. In other words, water and steel are only considered as the material of radiation sources for SC HX. The word “Vapor” described in Table 12.2-25 is a typographic error. This will be revised to correct word “Steel” as used in the shielding analysis. Refer to Attachment for the DCD markups.

The partial density 0.453 g/cm$^3$ for the steel is calculated as follows:

a. The geometric data of SC HX are as follows:

- HX active length (i.e. straight length of U-tube) : 803.148 cm
- Inner diameter of SC HX : 137.16 cm
- Outer diameter of tube : 1.905 cm
- Tube wall thickness : 0.0889 cm
- Quantity of tubes: # 840

b. The total volume of the source region is,

$$\pi\times\left(\frac{137.16\ \text{cm}}{2}\right)^2\times 803.148\ \text{cm}=1.187\text{E}+07\ \text{cm}^3$$

c. The volume of tube wall material (excluding the round part) is,

$$\pi\times\left\{\left(\frac{1.905\ \text{cm}}{2}\right)^2-\left(\frac{1.905\ \text{cm}-2\times 0.0889\ \text{cm}}{2}\right)^2\right\}\times 803.148\ \text{cm}\times 2\times 840=6.844\text{E}+05\ \text{cm}^3$$

d. The density of steel is assumed to be 7.86 g/cm$^3$. Therefore, the partial density of tube wall material in the source region of SC HX is,

$$\rho_{tube}=7.86\,\frac{\text{g}}{\text{cm}^3}\times\frac{6.844\text{E}+05\ \text{cm}^3}{1.187\text{E}+07\ \text{cm}^3}=0.453\ \text{g/cm}^3$$

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**Impact on DCD**

DCD Table 12.2-25 will be revised as indicated in the Attachment.

**Impact on PRA**

There is no impact on the PRA.

**Impact on Technical Specifications**

There is no impact on the Technical Specifications.

**Impact on Technical/Topical/Environmental Reports**

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

Table 12.2-25 (2 of 3)

| Building | Component | Source Dimension | | | | Source Characteristic | | Housing | |
|---|---|---|---|---|---|---|---|---|---|
| | | Shape | Diameter (or Width) (cm) | Length (cm) | Height (cm) | Material | Partial Density ($g/cm^3$) | Material | Thickness (cm) |
| Auxiliary Building | CS miniflow HX | Cylinder | 31.75 | - | 186.06 | Water: 94 %<br>Steel: 6 % | 0.94<br>0.45 | Steel | 0.95 |
| | Equipment drain tank | Cylinder | 193.59 | - | 610.87 | Water: 50 %<br>Vapor: 50 % | 1.00<br>0.001293 | Not considered | |
| | Boric acid concentrator | Cylinder | Liquid: 193.53<br>Vapor: 206.58 | - | 180.52 | Water: 47 %<br>Vapor: 53 % | 1.00<br>0.001293 | Not considered | |
| | SC HX | Cylinder | 137.16 | - | 803.15 | Water: 94 %<br>~~Vapor~~: 6 %<br>Steel | 0.942<br>0.453 | Steel | 1.27 |
| | SFP cleanup demin. | Cylinder | 145.70 | - | 144.17 | Water: 100 % | 1.00 | Not considered | |
| | Boric acid condensate IX | Cylinder | 74.60 | - | 206.17 | Water: 100 % | 1.00 | Not considered | |
| | Deborating IX | Cylinder | 105.08 | - | 104.49 | Water: 100 % | 1.00 | Not considered | |
| | Pre-holdup IX | Cylinder | 52.54 | - | 104.49 | Water: 100 % | 1.00 | Not considered | |
| | Purification IX | Cylinder | 52.54 | - | 104.49 | Water: 100 % | 1.00 | Not considered | |
| | SFP cooling HX | Rectangular parallelepiped | 31.19 | 134.16 | 198.28 | Water: 67 %<br>Steel: 33 % | 0.67<br>2.63 | Not considered | |
| | Volume control tank | Cylinder | 120.72 | | 218.09 | Water: 40 %<br>Vapor: 60 % | 1.00<br>0.001293 | Not considered | |
| | SGBD flash tank | Cylinder | 152.40 | - | 455.96 | Water: 100 % | 1.00 | Not considered | |
| | SGBD HX | Cylinder | 42.43 | - | 487.68 | Water: 86 %<br>Steel: 14 % | 0.90<br>1.12 | Steel | 1.27 |