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.:	**235-8275**
SRP Section:	**12.03 – 12.04 Radiation Protection Design Features**
Application Section:	**12.3 – 12.4**
Date of RAI Issue:	**10/07/2015**

Question No. 12.03-25

10 CFR 20.1101(b) requires that the licensee use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).

FSAR Section 12.3.1.2 indicates that tanks are provided with vents to facilitate the removal of potentially radioactive gases, however, the FSAR does not indicate where the vent lines send the potentially radioactive gases or if there are appropriate provisions to ensure the proper operation of the ventilation system.

1. Please update FSAR Section 12.3.1.2 to indicate where the tank vent lines send the potentially radioactive gases.

2. If gases are vented to the plant atmosphere, please indicate the vicinity of the vent line discharge points to the HVAC system intake and how the design minimizes airborne radioactivity in the plant.

3. Update the FSAR to discuss any provisions to prevent intrusion of liquids or moisture into the ventilation systems.

Response

As described in DCD Subsection 12.3.1.2, the APR1400 component design for system tanks includes vent lines for the removal of potentially radioactive gases, which are discharged within close proximity to the respective building ventilation intakes for collection and treatment.

1. DCD Subsection 12.3.1.2 will be updated as indicated below to describe the discharge location of the tank vents to be within close proximity to the building HVAC system intake point.

2. The tank vents discharge to the plant atmosphere within the respective cubicle in which the tank is located. The discharge point is located within close proximity to the building radioactive HVAC system intake for collection and treatment of potentially radioactive gases.

   The APR1400 features included in the plant layout and system design to minimize the spread of contamination are described in Subsection 12.3.1.4 for general airborne contamination control throughout the plant facilities and in Subsection 9.4.8 for the specific HVAC system design. These features are incorporated into the plant layout and HVAC systems design in order to minimize the spread of contamination throughout the plant in accordance with the ALARA principle as detailed in 10 CFR 20.1101(b) and RG 8.8 as well as with the design approaches described in RG 4.21 for contamination control.

   The primary features of the building radioactive HVAC systems and ventilation design include the airflow direction from areas of lesser contamination to higher contamination areas. The HVAC systems include component redundancy to ensure continuous system operation and the processing of potentially contaminated air flow. The potentially contaminated airflows are processed through filter trains, consisting of HEPA filters and charcoal filters for the removal of airborne radiological contaminants before the filtered exhaust stream is released via a monitored pathway to the plant vent ensuring that discharge limits are met.

3. The APR1400 tank design includes provisions for the prevention of the release of liquids via the tank vent line. Each tank design includes a vent line at the top of the tank above the overflow line for liquids, which is in turn located above the maximum fill level for liquid tank contents. The vent line terminates at the close proximity of the cubicle HVAC exhaust intakes, but a gap is maintained between the vent line discharge and the HVAC intakes to prevent liquid carry-over into the ventilation system. Tanks are provided with internal filters on the vent lines to prevent the release of liquids

   In the event that liquid exits the tank via the tank vent line, the liquid is released to the cubicle floor for collection via the drainage collection lines and building sumps.

   The building HVAC systems include provisions to handle moisture within the ventilation air stream. The HVAC ACUs include moisture separators to remove moisture from the air stream before it passes through the HEPA and charcoal filters for radionuclide removal. The moisture is collected and routed directly to the drain hub in the air cleaning unit. The drains are collected in the sumps and routed to LWMS for treatment and release.

Therefore, through the tank design as well as the HVAC systems design, the spread of contamination through the plant via the ventilation systems is minimized resulting in less radiation exposure to workers occupying plant areas.

Impact on DCD

DCD Subsection 12.3.1.2 Item d, 4 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 Report.

5) Cartridge filter housing closure heads are designed to swing free for the unobstructed removal of the cartridge.

6) Design features to handle spent filters are described in Subsection 11.4.2.

d. Tanks

1) Tanks are designed for complete drainage and are therefore free of internal crevices and pockets. The drain line is connected to the bottom of the tank.

2) Tanks are provided with at least one of the following means of decontaminating the tank internals based on tank contents and radioactivity levels:

a) Ample space to permit decontamination of the tank manway

b) Internal spray nozzles on potentially highly contaminated tanks for internal decontamination

c) Backflush capability for tank inlet screens

3) Tanks are designed with ellipsoidal or sloped bottoms to facilitate drainage and minimize the accumulation of crud.

4) ~~Tanks are provided with vents to facilitate the removal of potentially radioactive gases.~~

Tanks are provided with vents to facilitate the removal of potentially radioactive gases. The tanks are vented to the cubicle atmosphere within close proximity to the respective radioactive building HVAC system intake for collection and treatment of radioactive gases. Tanks are provided with internal filters on the vent lines to prevent the release of liquids

5) Non-pressu … oor drain pump or ot … ive fluids onto the flo … ed to the LWMS for

6) All tanks c … bilities to prevent settling of radioactive solids.

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.:	**235-8275**
SRP Section:	**12.03-12.04 – Radiation Protection Design Features**
Application Section:	**12.3-12.4**
Date of RAI Issue:	**10/07/2015**

Question No. 12.03-26

10 CFR 20.1101(b) requires licensees to control external occupational exposure, and to ensure that engineering controls are used to keep occupational doses ALARA. In 10 CFR 20, the definition for ALARA includes guidance to make every reasonable effort to maintain exposures below regulatory limits, taking into account the state of technology. 10 CFR 50 GDC 61 requires licensees to ensure that there is adequate shielding for routine activities in the area of the equipment.

While FSAR Sections 9.1.4 and 12.2.1.1.6 provide some information on in-core instrumentation (ICI), including the use of a hydraulic cutter to cut used ICI leads for disposal, and FSAR Section 12.3.2.3 discusses design features to limit access to the ICI chase, insufficient information is provided in the FSAR describing the insertion and removal process of in-core instrumentation. For example, the application does not clearly describe equipment used to insert and remove the in-core instruments. In addition, Industry experience shows that detectors can stick in the core, and require personnel to work at the location of the movable drive system to rectify the problem. There have been personnel over exposures due to working on in core detector systems. Insufficient information is available to determine the adequacy of the shielding and personnel protective features provided for the removal of in-core instrumentation.

1. Please discuss, in more detail, the process of removing and inserting in-core instrumentation from the core and preparing and transporting irradiated in-core instrumentation for storage and/or disposal and the equipment relied on to do so, including description of an underwater vacuum system to clean up debris from cutting ICI. If there is not an underwater vacuum system, discuss why this is acceptable.

2. In accordance with 10 CFR 20.1101(b), please update FSAR Section 12.3 to describe the personnel protective features and shielding provided for work on the ICI during an anticipated operational occurrence, such as stuck detectors and how the design ensures that doses to workers will remain ALARA during such an event.

Response

1. In-Core Instrumentation (ICI) activities are divided into three parts: A - withdrawal, B - disposal, and C - insertion) to be performed during refueling outage. These activities are generally performed remotely by personnel staying outside the water pool for radiation ALARA purpose, and are guided by operating procedures to minimize radiation exposure.

   A. Process for withdrawing ICI from the core:

      1) Disconnect MI cable from ICI assembly and from Pool Wall Junction Panel (PWJP).

      2) Remove MI cable tray assembly to a temporary storage area.

      3) Install ICI holding frame.

      4) Connect ICI retrieval tool to every ICI assembly.

      5) Withdraw ICI assembly using the hoist as In and Outward sequence.

      6) Position ICI assembly on ICI holding frame.

      This process is performed after the reactor area is partially flooded, but prior to opening of the reactor head, and locally near the ICI seal table. The ICI table seal area is to be surveyed for radiation level before entry to this area for performance of withdrawal work.

   B. Process for disposal of a used ICI assembly

      1) Withdraw a selected ICI assembly using the reel installed on the CEA change platform.

      2) Suspend the withdrawal activity when the guide tube is about 2 feet from the seal housing in order to grab the ICI assembly with the air operated plier on the CEACP to facilitate cutting of the ICI assembly for disposal.

      3) Grab the ICI assembly using air operated plier assembler

      4) Withdraw the ICI assembly completely.

      5) Direct ICI assembly into the funnel of ICI cutter on top of the transport container.

      6) Use hydraulic ICI cutter to shear ICI assembly into smaller pieces to fit into the transport container.

      7) Transport CEA/ICI transport container to fuel handling area (Fuel Building).

      8) Clean up the pool floor thoroughly after refueling operations are completed.

The process is performed remotely with operator staying further away from the ICI seal table and after the reactor area is fully flooded for refueling. Please note that the shearing operation uses a remotely operated hydraulic ICI cutter tool directly above the debris collection funnel so that debris from cutting the ICI assembly falls into the CEA/ICI transport container. This method minimizes the spread of chips and debris, and also minimizes radiation exposure from debris cleanup. Hence, underwater vacuum system is not required to be used.

C. Process for inserting ICI into the guide tubes

1) Hook ICI assembly to the hoist on the CEA/ICI change platform.

2) Insert ICI assembly into the guide tube as far as possible with insertion hand tool.

3) Install O-rings and backup rings on the seal plug as directed by the procedures.

4) Push down using the secondary insertion tool until the ICI assembly reaches to the top of the fuel.

5) Determine the number and size of spacers using a positioning gauge.

6) Remove the ICI holding frame.

7) Insert new ICI assembly into an unoccupied guide tube like same as Step C.2 above.

8) Put ICI MI cable tray assembly back in position.

9) Reconnect the MI cable to the ICI assembly and PWJP.

This process is performed locally near the ICI seal table.

2. DCD Subsection 12.3.2.3 will be updated to include the following paragraph.

"The APR1400 ICI design adopts the bottom mounted In-Core Instrument system that has only fixed type detectors for normal operation. During refueling operation, all ICIs are withdrawn from the core up to the full length of 24 feet when the reactor area is flooded with refueling water. The highly irradiated portion of the ICIs still remains inside the ICI guide tubes which are filled with refueling water at the lower head area of the reactor vessel. The withdrawal and disposal operation of ICI is performed remotely and the operators are shielded by the pool of water as well as the distance between the reactor vessel and the CEACP; and administrative procedures are used to monitor radiation level to insure personnel safety and radiation is ALARA."

Impact on DCD

DCD Subsection 12.3.2.3 will be updated as indicated in 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 Environment Report.

pipe chase. The resin transfer lines are also provided with a flushing capability to minimize the potential for hot spots in the piping.

~~The ICI chase is potentially a high-radiation area (greater than 1 Gy/hr) during ICI withdrawal. Stringent access control is provided to this area during movement of the ICI. A lockable access door is provided with a warning light. During withdrawal of the ICI, the warning light illuminates, providing indication that the ICI is being moved. An area radiation monitor is located in the ICI chase to provide indication of radiation levels and to alarm the personnel when the ICI is being withdrawn. Emergency egress from the area is also provided from the ICI chase.~~

Insert A(Next page)

Components that handle a significant amount of radioactive materials, such as LWMS floor drain tanks and equipment waste tanks, are located in shielded cubicles separated from the pump and valve galleries that are provided with labyrinths for access to the galleries. This design approach minimizes radiation streaming and scattering but permits inspection and maintenance access and removal of smaller items such as pumps, valves, and instruments for repair in lower-radiation areas. This design approach meets the requirements of NRC RG 8.8 2.b(4). The plant shielding is designed not only to maintain personnel occupational exposure ALARA, but also to maintain exposure to the general public ALARA.

The APR1400 shielding design has target dose rates that are below the limits for radiation zone designations provided in Table 12.3-2 to provide a sufficient margin in maintaining radiation exposure to plant personnel and the public ALARA.

12.3.3 Ventilation

The spread of airborne contamination within the plant is minimized by the design of the plant HVAC systems to provide airflow from areas of lower potential for airborne contamination to areas of greater potential for airborne contamination. For building compartments with the potential for contamination, the exhaust from the areas is designed with pressure and flow balances to minimize the amount of uncontrolled exfiltration from these areas. These design features provide reasonable assurance that the average concentration of radioactive material in the air in the areas that are normally occupied is less than the small fraction of DAC prescribed in 10 CFR Part 20 Appendix B. Therefore, A

The APR1400 ICI design adopts the bottom mounted In-Core Instrument system that has only fixed type detectors for normal operation. During refueling operation, all ICIs are withdrawn from the core up to the full length of 24 feet when the reactor area is flooded with refueling water. The highly irradiated portion of the ICIs still remains inside the ICI guide tubes which are filled with refueling water at the lower head area of the reactor vessel. The withdrawal and disposal operation of ICI is performed remotely and the operators are shielded by the pool of water as well as the distance between the reactor vessel and the CEACP; and administrative procedures are used to monitor radiation level to insure personnel safety and radiation is ALARA.

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.:	**235-8275**
SRP Section:	**12.03 – 12.04 Radiation Protection Design Features**
Application Section:	**12.3 – 12.4**
Date of RAI Issue:	**10/07/2015**

Question No. 12.03-31

10 CFR 100.11(a)(1) requires that the exclusion area boundary dose be limited to 25 rem in 2 hours following the onset of a postulated fission product release.

FSAR Section 12.3.2.1 Paragraph a.2.b. indicates that shielding is adequate to keep direct and scattered radiation below the 100.11(a)(1) limit (which is referenced in 10 CFR 50.34) and the limits in Chapter 15 during an accident. However, the 25 rem limit is a dose limit from a combination of all sources. Please revise FSAR Section 12.3.2.1, Paragraph a.2.b. to specify that the shielding is adequate to ensure that the total radiation dose to the whole body does not exceed the dose requirements. If the shielding design is inadequate to ensure that the exclusion area dose will remain below 25 rem from all sources, during the worst case design basis accident, increase plant shielding thicknesses, as appropriate.

Response

The TEDE dose limit of 250 mSv in 10CFR50.34(a)(1)(ii) is applied to APR1400. As indicated by staff, the combination of all sources due to a postulated LOCA should not exceed 250 mSv TEDE. The radiological consequences of a large break LOCA is analyzed to be 204 mSv at the EAB (DCD Table 15.6.5-14), and that the calculated direct dose at a distance of 800 meters from the LOCA sources inside the containment is 2.6 mSv, the total dose at the EAB is 206.6 mSv. Hence, the shielding is considered adequate.

Impact on DCD

DCD section 12.3.2.1 Paragraph a.2.b 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 Environment Report.

so that, after a postulated DBA, radiation exposure in the MCR for the duration of the accident does not exceed the TEDE of 50 mSv, including dose contributions from ingress and egress of the MCR.

The radiation shielding protecting the MCR and associated areas is designed based on the anticipated radiation environment resulting from the postulated DBA.

b) Direct offsite doses

~~Adequate shielding is provided to limit accident exclusion area boundary doses due to direct and scattered radiation from contained sources within the plant to the limits specified in 10 CFR 50.34 (Reference 11) and Chapter 15 of the Standard Review Plan (Reference 12).~~

b. Seismic and safety classification

Structural walls are designed to meet seismic category requirements. Structural walls are designed as seismic Category I, II, or III depending on the particular design requirements other than the radiation protection requirements such as structural integrity and load-bearing capacity.

The primary shield, the shield walls for the MCR, and the shield walls for the SFP are examples of shield walls that are designed as seismic Category I.

c. Protection of equipment and structures

Adequate shielding is provided for the following purposes:

1) To limit radiation heating of structural concrete

2) To reduce neutron activation of equipment

3) To limit the radiation dose to equipment and materials

"Adequate shielding is provided to limit the total radiation doses to the whole body not to exceed the limits specified in 10 CFR 50.34 (Reference 11) and Chapter 15 of the Standard Review Plan (SRP) (Reference 12). The shielding is designed adequately that the radiation dose to an individual located at the exclusion area boundary for a duration of 2 hours during a postulated DBA does not exceed 25 rem."

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

Docket No. 52-046

RAI No.:	**235-8275**
SRP Section:	**12.03 – 12.04 Radiation Protection Design Features**
Application Section:	**12.3 – 12.4**
Date of RAI Issue:	**10/07/2015**

Question No. 12.03-32

Regulatory Guide 8.8, Section C.4.b, provides guidance on the types of portable radiation monitoring instruments that would likely be needed at a light-water reactor.

While the FSAR provides COL item 12.3(1) on portable instruments, the COL item only appears to be focused on portable instruments needed for post-accident conditions and does not reference the guidance on portable instruments provided in Regulatory Guide 8.8. Please update the COL item to include portable instruments that may be needed during normal operations and AOOs and reference the guidance in Regulatory Guide 8.8 in the COL item, or justify an alternative.

Response

Subsection 12.3.4 and COL item 12.3(1) will revised to reference the use of Regulatory Guide 8.8.

Impact on DCD

DCD Subsections 12.3.4 and 12.3.6, Table 1.8-2 will be updated as indicated in 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 Environment Report.

RAI 14-7858 - Question 12.03-4

RAI 235-8275 -Question 12.03-32

Table 1.8-2 (20 of 29)

Item No.	Description
COL 11.5(4)	The COL applicant is to prepare an ODCM that contains a description of the methodology and parameters for calculation of the offsite doses for the gaseous and liquid effluents. The COL applicant is to follow NEI 07-09A as an alternative to providing an offsite dose calculation manual.
COL 11.5(5)	The COL applicant is to provide analytical procedures and sensitivity for selected radioanalytical methods and types of sampling media for site-specific matter.
COL 11.5(6)	The COL applicant is to develop the calibration procedures in accordance with NRC RG 1.33 and 4.15.
COL 11.5(7)	The COL applicant is to develop detailed location and tubing installation and provide the sampling method including the sampling time to acquire representative sampling.
COL 11.5(8)	The COL applicant is to provide operational procedures and maintenance programs related to leak detection and contamination control.
COL 11.5(9)	The COL applicant is to develop a radiological and environmental monitoring program, taking into consideration local land use and census data in identifying all potential radiation exposure pathways. The COL applicant is to follow NEI 07-09A as an alternative to providing a radiological and environmental monitoring program.
COL 12.1(1)	The COL applicant is to provide the organizational structure to effectively implement the radiation protection policy, training, and reviews consistent with operational and maintenance requirements, while satisfying the applicable regulations and Regulatory Guides including NRC RGs 1.33, 1.8, 8.8, and 8.10.
COL 12.1(2)	The COL applicant is to describe the operational radiation protection program to provide reasonable assurance that occupational radiation exposures are ALARA.
COL 12.1(3)	The COL applicant is to describe how the plant follows the guidance provided in NRC RGs 8.2, 8.7, 8.9, 8.13, 8.15, 8.20, 8.25, 8.26, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36, and 8.38.
COL 12.2(1)	The COL applicant is to provide any additional contained radiation sources, such as instrument calibration radiation sources, that are not identified in Subsection 12.2.1.
COL 12.3(1) 2	The COL applicant is to provide portable instruments and the associated training and procedures in accordance with 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737.
COL 12.3(2) 3	The COL applicant is to determine the WARN and ALARM setpoints of the ARMS based on the site-specific conditions and operational requirements

as well as the guidelines of RG 8.8.

COL 12.3(1) The COL applicant is to provide the material composition and shielding properties of the following doors/hatches, and these thicknesses equivalent to the minimum required concrete shield thicknesses.
- Personnel Air Lock between Containment Annulus Area (100-C01) and Personnel Air Lock Entrance (100-A14A)
- Personnel Air Lock between Operating Area (156-C01) and Containment Entrance Area (156-A04B)
- Equipment Hatch between Operating Area (156-C01) and Equipment Hatch Access Room (156-A10A)
- Door between Equipment Hatch Access Room (156-A10A) and the building exterior
- Doors between Truck Bay (100-P08) and the building exterior

Also, the COL applicant is to provide the service life of these doors/hatches and perform periodic in-service inspection and maintenance for these doors/hatches to provide reasonable assurance of functionality throughout the life of the plant.

personnel exposure due to inhalation of and contact with airborne contamination is maintained ALARA.

Airborne radiation monitoring is provided for areas that are normally occupied and have a significant potential for airborne contamination. The monitors can detect the time-integrated change of the airborne radioactivity within 10 DAC-hours for the most limiting particulate and iodine species in each area.

Airborne radiation monitors are described further in Section 11.5. The locations of the process effluent radiation monitors are shown in Figure 11.5-2. The airborne radiation monitors are located upstream of the filters within the HVAC ventilation systems.

HVAC systems are described in Section 9.4.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The area radiation monitoring system (ARMS) supplements the personnel and area radiation survey provisions of the plant health physics program described in Section 12.5 and provides reasonable assurance of conformance with the personnel radiation protection requirements of 10 CFR 20, 10 CFR Part 50, 10 CFR Part 70 (Reference 19); the guidelines of NRC RGs 1.21 (Reference 20), 1.97, 8.2 (Reference 21), 8.25 (Reference 5), and 8.8 (Reference 1); and American National Standards Institute (ANSI) N13.1-1999 (Reference 22) and Institute of Electrical and Electronics Engineers (IEEE) Std. 497-2002 (Reference 23). The ARMS is in conformance with ANSI/ANS HPSSC-6.8.1 (Reference 24).

The process and effluent radiation monitoring system and sampling systems are described in Section 11.5. Portable instruments are also used as needed during normal operation in accordance with the guidelines of RG 8.8.

Portable instruments are used and the associated training and procedures are provided to accurately determine the airborne iodine concentration in areas within the facility where plant personnel could be present during an accident in accordance with the requirements of 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737. The COL applicant is to provide portable instruments and the associated training and procedures in accordance with 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737 as well as the guidelines of RG 8.8. (COL 12.3(~~1~~2)).

d. Containment air monitor

e. Fuel handling area HVAC exhaust monitor

f. Condenser vacuum vent exhaust monitor

g. Main control room air intake monitor

h. Compound building HVAC exhaust monitor

i. Miscellaneous process monitors

12.3.4.2.1 Design Objectives

The objectives of the airborne radioactivity monitors are presented in Subsection 11.5.1.1.

12.3.4.2.2 Location of Airborne Radioactivity Monitors

The criteria for the location of the airborne radioactivity monitors are presented in Subsection 11.5.1.1 and the monitor locations are shown in Figure 11.5-2.

12.3.4.2.3 System Description

Airborne radioactivity monitors and applicable design criteria are described in Subsection 11.5.1.2.

12.3.5 Dose Assessment

The dose assessment is described in Section 12.4.

Insert "B" in next page.

12.3.6 Combined License Information

COL 12.3(~~1~~) 2 The COL applicant is to provide portable instruments and the associated training and procedures in accordance with 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737 as well as the guidelines of RG 8.8.

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SRP Section:	**12.03 – 12.04 Radiation Protection Design Features**
Application Section:	**12.3 – 12.4**
Date of RAI Issue:	**10/07/2015**

Question No. 12.03-33

10 CFR 50, GDC 63 requires that an appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

COL information item 12.3(2) indicates that the COL applicant is to determine the WARN and ALARM setpoints of the area radiation monitoring system based on the site-specific conditions and operational requirements. However, in addition to alarming, the containment operating area and containment upper operating area monitors provide an ESF function of sending a containment purge isolation actuation signal and the spent fuel pool area monitors provide an ESF function of sending a fuel handling area emergency ventilation actuation signal. Please update COL item 12.3(2) to indicate that the COL applicant will also determine the setpoints for these ESF actuation signals.

Response

The ALARM setpoints for the area radiation monitors send a signal to initiate an alarm in the MCR and local areas. The ALARM signal for the area radiation monitors for the containment operating area (RE-231A and RE-232B) and the containment upper operating area (RE-233A and RE-234B) also send a signal to initiate the containment purge isolation actuation signal (CPIAS), and the ALARM signal for the area radiation monitors for the spent fuel pool area (RE-241A and RE-242A) also send a signal to initiate the fuel handling area emergency ventilation actuation signal (FHAEVAS). COL 12.3(2) will be revised to include these functions.

Impact on DCD

DCD Tier 2 Table 1.8-2 and Sections 12.3.4.1.6 and 12.3.6 will be revised as indicated in 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 Environment Report.

Table 1.8-2 (20 of 29)

Item No.	Description
COL 11.5(4)	The COL applicant is to prepare an ODCM that contains a description of the methodology and parameters for calculation of the offsite doses for the gaseous and liquid effluents. The COL applicant is to follow NEI 07-09A as an alternative to providing an offsite dose calculation manual.
COL 11.5(5)	The COL applicant is to provide analytical procedures and sensitivity for selected radioanalytical methods and types of sampling media for site-specific matter.
COL 11.5(6)	The COL applicant is to develop the calibration procedures in accordance with NRC RG 1.33 and 4.15.
COL 11.5(7)	The COL applicant is to develop detailed location and tubing installation and provide the sampling method including the sampling time to acquire representative sampling.
COL 11.5(8)	The COL applicant is to provide operational procedures and maintenance programs related to leak detection and contamination control.
COL 11.5(9)	The COL applicant is to develop a radiological and environmental monitoring program, taking into consideration local land use and census data in identifying all potential radiation exposure pathways. The COL applicant is to follow NEI 07-09A as an alternative to providing a radiological and environmental monitoring program.
COL 12.1(1)	The COL applicant is to provide the organizational structure to effectively implement the radiation protection policy, training, and reviews consistent with operational and maintenance requirements, while satisfying the applicable regulations and Regulatory Guides including NRC RGs 1.33, 1.8, 8.8, and 8.10.
COL 12.1(2)	The COL applicant is to describe the operational radiation protection program to provide reasonable assurance that occupational radiation exposures are ALARA.
COL 12.1(3)	The COL applicant is to describe how the plant follows the guidance provided in NRC RGs 8.2, 8.7, 8.9, 8.13, 8.15, 8.20, 8.25, 8.26, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36, and 8.38.
COL 12.2(1)	The COL applicant is to provide any additional contained radiation sources, such as instrument calibration radiation sources, that are not identified in Subsection 12.2.1.
COL 12.3(1)	The COL applicant is to provide portable instruments and the associated training and procedures in accordance with 10 CFR 50.34(f)(2)(xxvii) and the criteria in Item III.D.3.3 of NUREG-0737.
COL 12.3(2)	The COL applicant is to determine ~~the WARN and ALARM setpoints of the ARMS~~ based on the site-specific conditions and operational requirements

the ARM setpoints for WARN, ALARM, and the containment purge isolation and fuel handling area emergency ventilation actuation signals, uncertainty calculations associated with the setpoints used for ESF actuation functions. The setpoint methodology follows the methodology in ANSI/ISA-67.04-1994 (Reference 27). The COL applicant is to determine ~~the WARN and ALARM setpoints of the ARMS~~ the ARM setpoints for WARN, ALARM, and the containment purge isolation and fuel handling area emergency ventilation actuation signals, based on the site-specific conditions and operational requirements (COL 12.3(2)).

12.3.4.1.7 Calibration Methods and Frequency

The methodology to determine the calibration methods and frequency of the ARMS is provided by the ODCM based on plant procedures.

12.3.4.1.8 Power Supplies

Instrument loops of safety-related monitors are powered from the appropriate train of Class 1E 120 AC distribution panel in the instrument power system (IP), which is powered by the onsite Class 1E emergency diesel generator. When the emergency diesel generator restores power to the skid, skid equipment such as sample pumps returns to the original operating status without having to be manually restarted. The TSC area radiation monitor, which is non-safety-related, is powered from permanent non-safety buses that are backed up by an alternate ac generator. Instrumentation and control power are described further in Subsection 8.3.2.

12.3.4.2 Airborne Radioactivity Monitoring Instrumentation

Airborne radioactivity monitors are installed in selected areas and HVAC systems to provide plant operating personnel with continuous information on the airborne radioactivity levels throughout the plant. These monitors, consisting of gaseous process and effluent radiation monitors (PERMS), are described in Section 11.5 and listed in Table 11.5-1. The airborne radioactivity monitors are as follows:

a. High-energy line break area HVAC exhaust monitor

b. Auxiliary building controlled area common HVAC exhaust monitor

c. Containment purge exhaust monitor

COL 12.3(2) The COL applicant is to determine ~~the WARN and ALARM setpoints of the ARMS~~ based on the site-specific conditions and operational requirements.

the ARM setpoints for WARN, ALARM, and the containment purge isolation and fuel handling area emergency ventilation actuation signals,

12.3.7 References

1. Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be ALARA," Rev. 3, U.S. Nuclear Regulatory Commission, June 1978.

2. Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable," Rev. 1-R, U.S. Nuclear Regulatory Commission, May 1977.

3. 10 CFR Part 20, "Standards for Protection against Radiation," U.S. Nuclear Regulatory Commission.

4. Regulatory Guide 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," U.S. Nuclear Regulatory Commission, June 2008.

5. Regulatory Guide 8.25, "Air Sampling in the Workplace," Rev. 1, U.S. Nuclear Regulatory Commission, June 1992.

6. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," U.S. Nuclear Regulatory Commission.

7. NUREG-0737, "Clarification of TMI Action Plan Requirements" U.S. Nuclear Regulatory Commission.

8. Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Rev. 4, U.S. Nuclear Regulatory Commission, June 2006.

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

10. Regulatory Guide 1.69, "Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Rev. 1, May 2009.

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.:	**235-8275**
SRP Section:	**12.03 – 12.04 Radiation Protection Design Features**
Application Section:	**12.3 – 12.4**
Date of RAI Issue:	**10/07/2015**

Question No. 12.03-41

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.

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 12.3-12.4 indicates that the plant shielding design and normal operation radiation zoning should consider conditions of normal operation, refueling, and anticipated operational occurrences (AOOs), including fuel handling and storage and radioactive material handling, processing, use, storage, and disposal.

ISSUE

FSAR Table 12.3-4 provides design basis minimum radiation shield thicknesses for the plant. However, while in the response to RAI 7930, Question 12.03-6, the applicant provided shielding information for the refueling canal (room 119-A01B), the applicant does not provide clear shielding information for the fuel transfer tube. In addition, it is unclear if the radiation zones account for radiation dose rates during fuel transfer.

Information Needed

1. Please update FSAR Table 12.3-4 to provide the minimum radiation shield thicknesses for the fuel transfer tube based on transferring of the maximum source term (maximum two fuel assemblies at the earliest time transfer would be allowed by technical specifications) that could be contained within the tube.

2. Please ensure that zoning for the areas surrounding the fuel transfer tube in FSAR Figures 12.3-4 through 12.3-8 include dose contributions from the maximum source term that would be expected within the fuel transfer tube during fuel transfer operations (considering the required minimum shielding to be provided in FSAR Table 12.3-4).

Response

1. The fuel transfer tube area is surrounded by the plant north and south walls, while the west side is adjoining to the containment, and the east side is connected to the refueling canal. The ceiling above is a controlled access area leading to a pipe chase. The floor below the transfer tube is another pipe chase. Please refer to Figures 1 and 2 for locations of these cubicles. The radiation zonings during normal power operation are as follows;

    - Cubicle beyond north wall is the mechanical penetration room (cubicle 120-A16B): Zone 6
    - Cubicle beyond south wall is another mechanical penetration room (cubicle 120-A16A): Zone 6
    - West end is connected to the refueling pool area (cubicle 130-C01) inside containment building: Zone 8 during power and refueling operations
    - East end is connected to the refueling canal (cubicle 119-A01B) Zone 2 during power operation and Zone 8 during refueling operations
    - Cubicle immediately above ceiling is the fuel transfer tube inspection area at 113'-10" elevation (the cubicle number is not currently provided but is below cubicle 137-A40B) are Zone 2 during power operation and Zone 3 during refueling operation
    - Cubicle below transfer tube floor is a pipe chase (078-A21B): Zone 6

    The minimum radiation shield wall thicknesses for the transfer tube are determined based on the maximum source terms of two spent fuel assemblies at the earliest time during transfer. The wall thicknesses are summarized as follows and will be added into Table 12.3-4 accordingly.

Table 1 Minimum Required Shield Thickness for Fuel Transfer Tube Area

Room Number	Room Name	Minimum Required Shield Thickness (inches)					
		North	South	East	West	Floor	Ceiling
-	Fuel Transfer Tube Area	44	44	No walls		62	60

It is noted that the transfer tube inspection area at Elevation 113'-10" is not to be accessed during refueling operations. This area can only be accessed through a hatch from cubicle 137-A40B. Cubicle 137-A40B is equipped with administratively locked doors and will have radiation signage posted to prevent inadvertent entry.

TS

Figure 1 The Location of the Fuel Transfer Tube Area (El. 120’)

TS

Figure 2 The Location of the Transfer Tube Inspection Area (El. 137'-6")

2. As discussed in item #1 above, the transfer tube inspection area at Elevation 137'-6" will be increased to zone 3 during refueling operations. A note to restrict entrance to this area will be added in Figure 12.3-6.

Impact on DCD

Table 12.3-4 and Figure 12.3-6 will be updated as indicated in Attachment 1 and 2.

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 Environment Report.

RAI 141-8098 - Question 12.03-08

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"A"

Table 12.3-4 (4 of 7)

Room Number	Room Name	Minimum Required Shield Thickness (inches)					
		North	South	East	West	Floor	Ceiling
Auxiliary Building (cont.)							
078-A37A	Deborating IX Room	15	12	10	24	24	24
078-A38A	SFP Cleanup Pump Room	23	23	23	10	23	32
078-A39A	Gas Stripper Effluent Radiation Monitor Room	23	23	23	10	16	35
078-A40B	Boric Acid Concentrator Room	16	23	23	14	16	16
086-A01A	Filter Area	-	-	18	21	13	10
100-A32B	SFP Cooling HX Room	10	10	10	10	10	10
100-A29B	Pipe and HVAC Chase	10	12	10	66	10	10
100-A13A	Mechanical Penetration Room	48	48	48	48	34	13
100-A13B	Mechanical Penetration Room	48	10	48	48	36	10
100-A16D	Pipe Chase	48	48	48	48	10	23
100-A16C	Pipe Chase	48	48	48	48	13	10
100-A24A	SFP Cooling HX Room	12	10	12	40	24	10
100-A26A	Valve Room	28	41	21	28	32	10
100-A25A	Volume Control Tank Room	42	42	42	47	48	53
111-A01B	Cask Loading Pit	48	14	48	48	42	-
114-A01B	Spent Fuel Pool	62	60	59	68	71	-
119-A01B	Refueling Canal	60	59	62	48	62	-
120-A16B	Mechanical Penetration Room	29	27	33	48	18	29
120-A16A	Mechanical Penetration Room	20	24	20	48	17	19
120-A23A	Valve Room	18	25	18	18	10	18
120-A14A	SG Blowdown Regen. HX Room	12	10	10	21	14	21
137-A19A	SG Blowdown Flash Tank Room	18	18	18	21	18	18
156-A14A	Aux. Bldg Controlled Area (I) Normal Exhaust ACU Room	18	18	18	18	18	18
174-A15B	Containment High- and Low-volume Purge ACU Room	21	21	21	21	15	10
195-A08B	Aux. Bldg. Controlled Area (II) Normal Exhaust ACU Room	18	18	18	18	18	18

Insert

-	Fuel Transfer Tube Area	44	44	No walls	62	60

Security-Related Information – Withheld Under 10 CFR 2.390

Figure 12.3-6 Radiation Zones(Normal) Auxiliary/Containment Building El.137'-6"