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.:	**390-8479**
SRP Section:	**12.02 – Radiation Sources**
Application Section:	**12.02**
Date of RAI Issue:	**02/01/2016**

Question No. 12.02-26

This is a follow-up to RAI 8247, Question 12.02-16.

REGULATIONS AND GUIDANCE

10 CFR 52.47(a)(8) requires that the FSAR contain, the information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v).

10 CFR 50.34(f)(2)(vii) requires that the applicant preform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment and references NUREG-0737, Section II.B.2.

10 CFR 50, GDC 19 requires that radiation exposure for the duration of an accident does not exceed 5 rem whole body, or its equivalent to any part of the body.

NUREG-0737, Section II.B.2, provides the systems that should be assumed in the post-accident shielding analysis to contain high levels of radioactivity.

SRP 12.3-12.4 indicates that the staff will conduct shielding design review to ensure that the design permits adequate access to important areas and provides for protection of safety equipment from radiation, following an accident. In addition, SRP 12.312.4 indicates that the staff will review the basis for the radiation shielding design.

INFORMATION NEEDED

In the response to RAI 8247, Question 12.02-16, the applicant provided a calculation for the dose in the control room envelope (CRE) from the MCR emergency filters and provides other information on accident source terms and the dose inside the CRE. Based on the response, staff has the following questions.

1. NUREG-0737, Section II.B.2 indicates that the systems that should contain high levels of radioactivity in post-accident situation should include, the chemical and volume control system, sample lines, gaseous radwaste systems, and standby gas treatment systems, along with any others and that if any of these systems were excluded, the applicant should explain why such systems were excluded.

   However, the applicant did not provide a post-accident source term for the above systems or provide any justification for why the post-accident source terms for the above systems were not included. Likewise, it does not appear that the possible post-accident source terms of the above systems were included in the dose analysis for post-accident access to vital areas.

   Please provide the potential post-accident source term for the systems listed above and any other applicable systems that may contain post-accident source term, considering all possible design basis accidents, and consider the dose rate from these areas in the vital area access analysis. If it is not necessary to include the post-accident source term for any of these systems, please provide justification for why a post-accident source term and analysis is not needed for these systems.

2. In the response, the applicant models the dose calculation from the MCR emergency filters to the CRE. However, neither the application nor the response provides the density of concrete that was assumed for calculating the accident dose inside the CRE. In addition, the assumed density of concrete was not provided in the application for any of the concrete walls used for radiation shielding. Please specify the density of concrete that was assumed in the radiation shielding calculations, including shielding for the CRE and all other radiation shields and update the FSAR to provide this information.

3. In the response, the applicant provides information indicating that containment penetrations will not be a significant dose contribution to CRE dose. Please specify if the design is such that it prevents a direct radiation streaming path or near direct radiation streaming path from the emergency filters (located above the main control room.) into the CRE through the ventilation ducting openings or other penetrations into the CRE.

<u>Response</u>

1. Among the several design basis accidents (DBAs) discussed in DCD Tier 2 Chapter 15, the loss of coolant accident (LOCA), postulating a piping break at the cold legs of the reactor coolant pumps (RCPs) is found to be most limiting case that may have impacts on the chemical and volume control system (CVCS), primary sampling lines (PXS), gaseous radwaste system (GRS), and standby gas treatment systems. KHNP evaluated the impacts of a LOCA on these systems as follows:

   - CVCS

   The CVCS is designed to provide treatment of reactor coolant through the letdown process for the control of primary coolant chemistry. The letdown heat exchangers, the purification filters and the ion exchangers are designed accordingly to remove radionuclides from the coolant. During a LOCA, the coolant radioactivity remains unchanged. Hence the normal reactor coolant source terms bound the design, including the post-accident LOCA source terms.

- PXS

The shielding design for the primary off-gas sampling and post-accident sampling rooms incorporates the DBA LOCA source terms. The source term information is summarized in shielding calculation #1-321-N376-014. The shielding calculation contains analysis of the gaseous and liquid piping in these rooms and the DBA source terms are summarized in Tables III-1 and III-2 for gaseous and liquid nuclides respectively, in the calculation. This calculation was provided to the NRC for the shielding calculation audit.

- GRS

The GRS is designed to take noble and halogen gases from the reactor coolant in the volume control tank and the reactor coolant drain tank. From the onset of the LOCA, there is no additional reactor coolant inputs to these tanks. Hence, the GRS is not anticipated to receive additional noble and halogen gases from these tanks. Therefore the GRS is not impacted by the LOCA and a post-accident source term and analysis is not needed for the GRS.

- Standby Gas Treatment Systems

As the terminology of standby gas treatment system is not used in the APR1400 design, KHNP interprets this as relating to the containment building low volume purge exhaust ACU. Accordingly, the containment low volume purge system (CLVPS) continues to operate following the large-break LOCA before containment isolation is initiated. The isolation valves of the CLVPS are closed by the containment isolation actuation signal (CIAS) after a LOCA with a loss of on-site power (LOOP) within 5.0 seconds. The CLVPS release evaluation assumes that 100 percent of the radionuclide inventory in the RCS is released to the containment at the onset of a LOCA and homogeneously mixed in the containment atmosphere. However a release of gap activity into the containment is not considered because the CLVPS operation terminates within 5 seconds, which is before the gap release into the reactor coolant that would occur in about 30 seconds. Please refer to DCD Tier 2 Subsection 15.6.5.5.1.3 “Containment Low Volume Purge System Release” for additional discussion.

Based on the above, the CLVPS is not impacted by the LOCA and a post-accident source term is not needed.

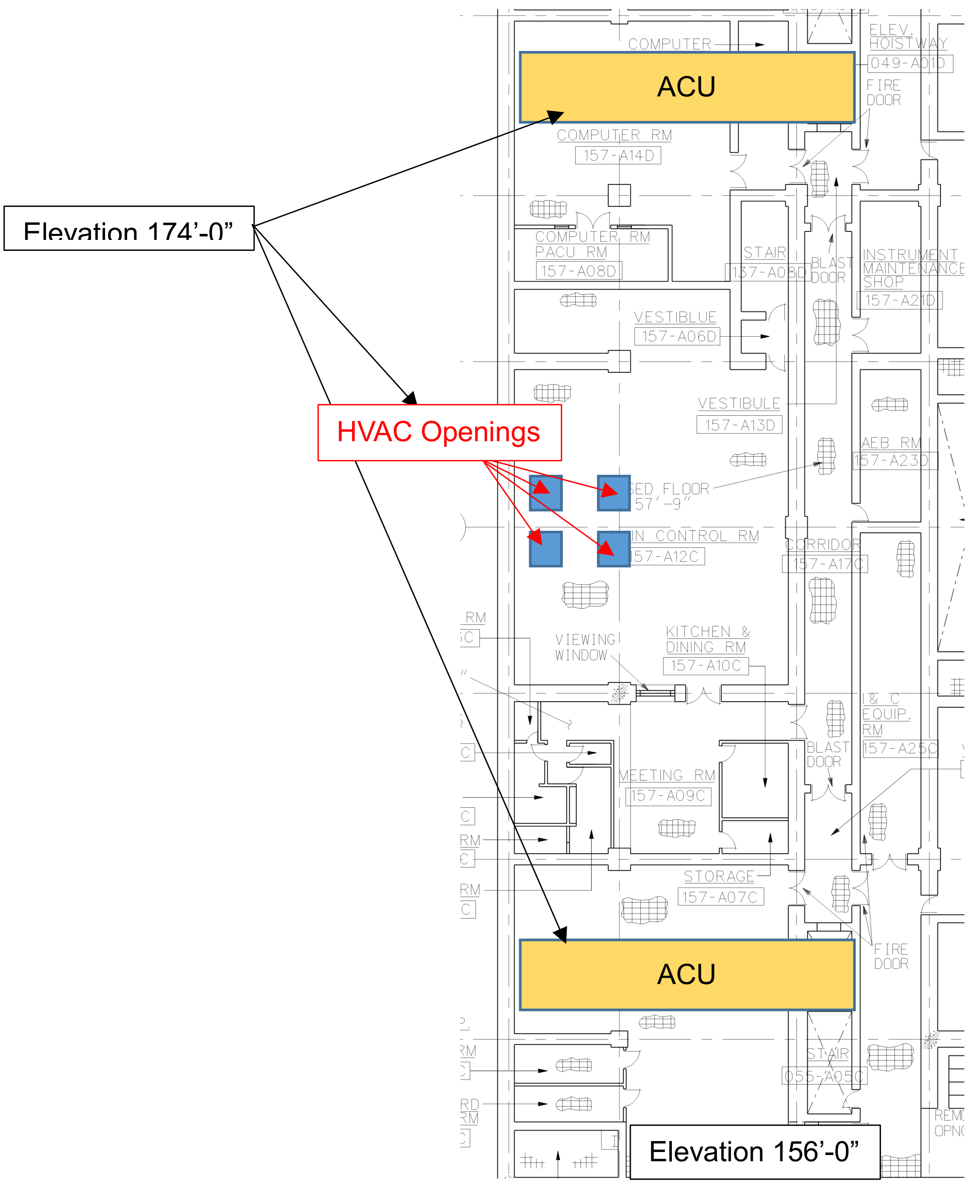
Please refer to the response to RAI No. 235-8275 Question No. 12.03-30 for a discussion on the post-accident vital areas and the exclusion of the CVCS for the vital areas. Based on the above discussion, no addition changes to the DCD are required.

2. The concrete density of 2.242 grams/cubic centimeter (140 pounds/cubic foot), which was used in all the shielding analyses will be provided in DCD Section 12.3.2.2 ‘Shielding Analysis’. KHNP believes that this concrete density is reasonably

conservative as an input to the shielding calculations in comparison to typical industry values.

3. The two control room emergency air cleaning units, at elevation 174'-0", are not located directly above the main control room (MCR) at elevation 156'-0"; one is located above the computer room (plant north side of the MCR), and the other above the TSC room on the south side of the MCR. Their locations are indicated in the attached sketch (Figure 1) for illustration purposes. Based on these locations, the air supply and exhaust duct penetrations are designed to be located away from the CR Emergency ACU to prevent direct, or near direct, radiation streaming pathways.

Figure 1. Location of CR Emergency ACUs above the MCR



In addition to the above responses, the following clarifications are being made in the DCD based on shielding calculation audit open items:

a). The following description of the shield plugs will be added in an appropriate location in DCD Section 12.3.2.3 'Shielding Design'.

'Two layers of shield plugs are installed around the reactor vessel. The shield plugs are made of ordinary *concrete and are designed to reduce the neutron and gamma streaming from the reactor as low as possible such that the dose rate on the operating floor is minimized. Adequate air cooling is provided to the area between the reactor and the shield plugs and the primary biological shield to ensure that the heat generated from the reactor does not cause any functional deterioration of the concrete with respect to the shielding and structural integrity.'*

b). The thickness for the east wall of 055-A22B for the post-accident sample control panel room and post-accident sample room is not consistent with the shielding requirement during post-accident conditions and will be changed from 27 inches to 48 inches.

Originally the minimum required shield thickness of 27 inches was determined to meet the zoning requirements for the normal operating condition. If this thickness were to be for an accident condition, the dose rate at the post-accident sample control room would be greater than 200 mrem/hr. However, the actual structure thickness of the wall is 48 inches. This wall thickness will allow for the dose rate in the room to be maintained low enough to perform the post-accident sampling activities, which is within the dose rate range of post-accident radiation Zone 2 (less than 100 mrem/hr). Therefore, the minimum shield thickness for this wall will be changed to 48 inches in DCD Table 12.3-4.

Impact on DCD

DCD Tier 2 Section 12.3.2.2, Section 12.3.2.3, and Table 12.3-4 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.

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.

d. Maintenance, inspection, and testing considerations

Adequate shielding is provided to provide reasonable assurance of safe personnel access and sufficient stay time near areas containing radioactive equipment for maintenance, inspection, and testing activities.

e. Additional requirements

The other shielding systems functional requirements generally depend on the location of the shield and the access requirements to or from the equipment or areas within the shield walls. Thus, access to an area may be through the shield itself such as through removable shield walls.

12.3.2.2 Shielding Analysis

The shielding analyses are based on a concrete density of 2.242 g/cm^3 (140 lbs/ft^3).

Calculations to determine the adequacy of the shielding design are based on the source strengths described in Subsection 12.2.1 and the methods described below. Dose points are selected for analysis inside and outside cubicles containing radioactive equipment. Skyshine from the cubicles is negligible because cubicles containing radioactive material are shielded overhead.

The only major source generating radionuclides is the reactor core at full power. The codes ANISN (Reference 13) and MCNP (Reference 14) are used to verify the effectiveness of the primary shield and evaluate the streaming of neutron and gamma radiation from the reactor vessel. Sources of gamma radiation are distributed throughout the reactor containment building and nuclear island. The codes MICROSHIELD (Reference 15) and RUNT-G (Reference 16) are used to verify gamma source shielding. The following sequence typifies a gamma source shielding analysis:

a. Determine the concentration of each principal nuclide in the source medium.

b. Adjust the concentration to account for issues such as accumulation, dilution, decay, and removal.

c. Convert the resulting concentrations into gamma source strength.

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

Insert A (next page)

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.

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

Two layers of shield plugs are installed around the reactor vessel. The shield plugs are made of ordinary concrete and are designed to reduce the neutron and gamma streaming from the reactor as low as possible such that the dose rate on the operating floor is minimized. Adequate air cooling is provided to the area between the reactor and the shield plugs and the primary biological shield to ensure that the heat generated from the reactor does not cause any functional deterioration of the concrete with respect to the shielding and structural integrity.

APR1400 DCD TIER 2

RAI 141-8098 - Question 12.03-08

RAI 390-8479 - Question 12.02-26

Table 12.3-4 (2 of 7)

Room Number	Room Name	Minimum Required Shield Thickness (inches)					
		North	South	East	West	Floor	Ceiling
Auxiliary Building (cont.)							
055-A21B	Pipe Chase and Valve Room	48	10	20	10	Ground	32
055-A22A	Pipe Chase	10	10	33	42	Ground	10
055-A22B	Pipe Chase	10	10	~~27~~ 48	42	Ground	10
055-A30A	SC HX Room	22	30	30	22	Ground	25
055-A30B	SC HX Room	30	30	30	22	Ground	28
055-A31B	Chemical Drain Sump Pump Room	27	10	10	10	Ground	20
055-A33A	Equipment Drain Sump Pump Room	10	22	14	10	Ground	14
055-A33B	Equipment Drain Sump Pump Room	19	10	10	10	Ground	17
055-A34A	Floor Drain Sump Pump Room	10	22	10	22	Ground	11
055-A34B	Floor Drain Sump Pump Room	22	10	10	22	Ground	18
055-A38A	Boronometer Room	25	25	34	25	Ground	28
055-A39A	Process Radiation Monitor Room	19	19	11	19	Ground	21
055-A42A	Charging Pump Room	17	24	24	28	Ground	22
055-A43A	Charging Pump Miniflow HX Room	10	18	10	10	Ground	19
055-A45A	Pipe Chase	27	17	25	10	Ground	24
055-A46A	Condensate Return Unit Room	28	37	10	16	Ground	16
055-A47B	Primary Off-Gas Sample Pump Room	16	10	16	15	Ground	10
055-A51B	Equipment Drain Tank Room	12	28	37	26	Ground	24
055-A52B	Reactor Drain Pump Room	13	12	10	13	Ground	17
055-A53B	Reactor Drain Pump Room	13	12	23	10	Ground	17
055-A54B	Aux. Charging Pump Room	10	14	14	12	Ground	13
055-A55B	Charging Pump Room	14	14	14	20	Ground	17
055-A56A	Valve Room	10	18	28	10	Ground	19
055-A56B	Valve Room	14	20	12	10	Ground	10
055-A58A	Pipe Chase	18	10	26	26	Ground	10
055-A59A	Valve Room	19	10	26	34	Ground	24